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Civilian Radioactive Waste Management System
Management and Operating Contractor

Waste Form Degradation and Radionuclide Mobilization Expert Elicitation Project

May 29, 1998

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Expert Elicitation Project

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1.0 INTRODUCTION

1.1 OBJECTIVES

This report presents results of the Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) project for Yucca Mountain, Nevada. This project was sponsored by the U.S. Department of Energy (DOE) and managed by Geomatrix Consultants, Inc. (Geomatrix), for TRW Environmental Safety Systems, Inc. The DOE's Yucca Mountain Site Characterization Project (referred to as the YMP) is intended to evaluate the suitability of the site for construction of a mined geologic repository for the permanent disposal of spent nuclear fuel and high-level radioactive waste. The WFEE project is one of several that involve the elicitation of experts to characterize the knowledge and uncertainties regarding key inputs to the Yucca Mountain Total System Performance Assessment (TSPA). The objective of the current project was to characterize the processes of degradation of spent fuel and high level waste (HLW) glass following breach of the waste packages and mobilization of radionuclides within breached waste packages. An understanding of these processes is critical to evaluating the performance of the potential high-level nuclear waste repository at Yucca Mountain.

Waste form degradation depends on many factors that are uncertain and variable, including (1) the chemical and physical conditions inside the breached waste packages (in particular, the composition of water that contacts the waste form); (2) the effect of cladding in minimizing the exposed surface area of spent nuclear fuel; (3) the amount of exposed surface areas and the surface area in contact with water, and (4) the development of secondary phases and the ability of these phases to modify solubilities. One concern relevant to modeling the waste form degradation is the applicability of short-term laboratory observations to in-situ conditions and extrapolation of the data to long-term behavior. The radionuclide mobilization will be affected by many factors that also are uncertain and variable, including (1) evolution of radionuclide-bearing secondary phases that control the concentration of the radionuclides in water; (2) physical and chemical

characteristics of degradation products of structural components of waste packages; and (3) the existence and role of colloids.

So that the analyses for the study included a wide range of perspectives, multiple individual judgments were elicited from members of an expert panel. The panel members, who were experts from within and outside the Yucca Mountain project, represented a range of experience and expertise. A deliberate process was followed in facilitating interactions among the experts, in training them to express their uncertainties, and in eliciting their interpretations. The resulting assessments, therefore, provide a reasonable aggregate representation of the knowledge and uncertainties about key issues regarding the waste form degradation and radionuclide mobilization at the Yucca Mountain site.

1.2 RELATIONSHIP OF WFEE PROJECT TO STUDIES FOR YMP

The WFEE project has two principal purposes: (1) to quantify uncertainties associated with certain key issues in the Total System Performance Assessment (TSPA); and (2) to provide a perspective on modeling and data collection activities that may help characterize and reduce uncertainties. The next iteration of the TSPA is being conducted for the Viability Assessment (VA) for Yucca Mountain. The TSPA-VA provides a probabilistic assessment of the performance of the potential repository based on the information developed through site characterization and repository design. The technical components of the TSPA are intended to incorporate a range of knowledge and uncertainties. As such, the expert panel's assessment of key technical issues related to waste form degradation and radionuclide mobilization—including the expressions of uncertainty in processes and models—will be directly applicable to the TSPA-VA.

In addition, the WFEE study is intended to complement the ongoing modeling, testing, and data collection programs (conducted principally by the national laboratories, including Lawrence Livermore National Laboratory [LLNL] and Argonne National Laboratory) while contributing to the performance assessment (conducted by the M&O).

The WFEE experts were given detailed summaries and presentations of available data, models, and the progress being made in various components of the modeling and testing program. The focus of the WFEE project was on evaluating the uncertainties associated with the various models, parameters, and processes and to provide a broad perspective on appropriate models to characterize waste form degradation mechanisms and rates. As such, the WFEE project is a logical step in the design and testing program for the Yucca Mountain project.

The Viability Assessment will rely largely on the next round of performance assessment (TSPA-VA). The TSPA-VA will be an assessment at a particular point in time of the level of knowledge and uncertainties regarding the site characteristics and engineered system that will affect performance of the potential repository system. As such, the performance assessment requires a reasonably complete description of all key processes affecting performance, including waste form degradation and radionuclide mobilization. Further, the TSPA, as a probabilistic analysis, will include appropriate expressions of uncertainties. The quantification of uncertainties at any given time does not imply that issues have been resolved, that additional data should not be gathered, or that the issues will not be revisited during subsequent evaluations (e.g., licensing). A goal of the WFEE project is to support the TSPA-VA by providing an expression of uncertainties regarding key issues. This expression has been developed by experts from both within and outside the YMP. As such, results of the WFEE study are realistic and defensible assessments at this point in the characterization program for Yucca Mountain. In addition to providing inputs to the TSPA-VA, the results of the study can also provide a focus for subsequent data collection aimed at reducing key uncertainties.

1.3 ORGANIZATION

WFEE project personnel were organized into four primary groups: the WFEE contractor, the methodology development team (MDT), the expert panel, and the technical specialists. The principal responsibilities of each group are described here; the technical roles of each group are described in detail in Section 2.2 of this report.

WFEE Contractor: Under contract with TRW, the WFEE contractor, Geomatrix, was responsible for conducting all aspects of the project and for delivering this report describing the methodology and the results. The WFEE contractor personnel also were members of the MDT.

Methodology Development Team (MDT): As a group, the MDT served both to carry out the project and to review its progress. Direct participation included developing a project plan, facilitating workshops, eliciting members of the expert panel, and documenting the methodology and results. The review role included reviewing the progress of the study and recommending mid-course adjustments to ensure that the study met its objectives. The members of the MDT and their responsibilities for the WFEE project are summarized in Table 1-1.

Expert Panel: The six widely recognized scientists on the expert panel were responsible for providing and documenting their judgments regarding models, parameters, and uncertainties about waste form degradation and radionuclide mobilization at Yucca Mountain. These subject-matter experts were responsible for developing the interpretations that form the technical substance of the WFEE project. Table 1-2 lists the experts on the panel and their affiliations. Brief biographies for members of the expert panel are provided in Appendix A.

Technical Specialists: Numerous technical specialists participated in the project by providing the experts with specialized data, interpretations, or training through workshops. A list of the technical specialists and their affiliations is given in Table 1-3. In some cases, members of both the MDT and the expert panel also acted as technical specialists.

1.4 PRODUCTS OF STUDY AND STRUCTURE OF REPORT

The WFEE study was conducted in approximately eight months. The project began with developing a plan for the course of the study and identifying the goals to be accomplished and methodologies to be implemented in meeting these goals. Next, the MDT developed and implemented a process for selecting the members of the expert panel, resulting in the selection of six experts. The bulk of the study was centered around three workshops. The workshops were designed to facilitate interaction among the experts, provide all data

needed for their assessments, and provide a forum for discussing a range of technical interpretations. Following the third workshop, the interpretations of each expert were elicited in individual interviews and documented in elicitation summaries. After reviewing the elicitation summaries of all members of the expert panel, the experts finalized their assessments.

This report contains the products of the WFEE project outlined above. Section 2 describes in detail the process followed in eliciting the expert interpretations. Appendices B and C provide summaries of the references and reports provided to the experts, and of the three workshops. This information provides written documentation of the technical data discussed by the panel, the formats and content of interpretations presented by outside technical specialists during the study, and the expert panel's preliminary interpretations.

Section 3 of this report presents in detail the final interpretations provided by the expert panel and the results of the study. Both the results for each of the six individual experts and the aggregated results are discussed. Key products of the study are the written elicitation summaries prepared by each expert, which are provided in Appendix D. The experts expended considerable effort to ensure that their summaries provide a reasonably complete record of the thought process they followed in arriving at their interpretations. Information related to quality assurance is provided in Appendix E.

TABLE 1-1
METHODOLOGY DEVELOPMENT TEAM MEMBERS AND THEIR PRINCIPAL RESPONSIBILITIES

NAME	AFFILIATION	RESPONSIBILITIES
Kevin J. Coppersmith	Geomatrix Consultants, Inc.	Project management and planning; methodology development; facilitating workshops; documentation
Roseanne C. Perman	Geomatrix Consultants, Inc.	Project planning and methodology development; organizing workshops; documentation
Robert R. Youngs	Geomatrix Consultants, Inc.	Project planning and methodology development
William Halsey	M&O/Lawrence Livermore National Laboratory	Project planning and methodology development; workshop planning; documentation
David Haught	U.S. Department of Energy	Project oversight
Eric R. Siegmann	M&O/Duke Engineering and Services	Project planning and methodology development; workshop planning
Peter A. Morris	Applied Decision Analysis	Project planning and methodology development; peer review of project direction; expert elicitation methodologies
Martha W. Pendleton	M&O/Woodward-Clyde Federal Services	Project planning and oversight; expert selection process; review of project direction
Ray Stout	M&O/Lawrence Livermore National Laboratory	Project planning and methodology development; workshop planning; documentation
Thomas Thornton	M&O/Framatome Comega Fuels	Project planning and methodology development; workshop planning

TABLE 1-2
EXPERT PANEL MEMBERS

EXPERT	AFFILIATION
Michael J. Apted	QuantiSCI
John K. Bates	Argonne National Laboratory
William Bourcier	M&O/Lawrence Livermore National Laboratory
Lawrence H. Johnson	Atomic Energy of Canada Limited
Henry F. Shaw	Lawrence Livermore National Laboratory
David W. Shoesmith	Atomic Energy of Canada Limited

TABLE 1-3
TECHNICAL SPECIALISTS PARTICIPATING IN WFEE WORKSHOPS *

WORKSHOP 1 – SIGNIFICANT ISSUES AND AVAILABLE DATA	
Robert Andrews	M&O/DESI
Lee Bendixsen	Idaho National Engineering Laboratory
Robert Einzinger	Pacific Northwest National Laboratory
Robert Finch	Argonne National Laboratory
Walt Gray	Pacific Northwest National Laboratory
Cynthia Palmer	M&O/Lawrence Livermore National Laboratory
Steve Steward	M&O/Lawrence Livermore National Laboratory
Ananda Wijesinghe	M&O/Lawrence Livermore National Laboratory

WORKSHOP 2 - ALTERNATIVE MODELS AND INTERPRETATIONS	
Tae Ahn	U.S. Nuclear Regulatory Commission
Yueting Chen	M&O/DESI
Richard Codell	U.S. Nuclear Regulatory Commission
David Franklin	Bettis Laboratory
Jerry McNeish	M&O/DESI
William Murphy	Center for Nuclear Waste Regulatory Analyses/Southwest Research Institute
David Sassani	M&O/DESI
John Tait	Atomic Energy of Canada Limited
Inez Triay	M&O/Los Alamos National Laboratory
David Wronkiewicz	University of Missouri

* Some members of the MDT and the expert panel also acted as technical specialists at the workshops; their names are not repeated here.

2.0 PROCESS FOR ELICITING EXPERT JUDGMENTS

2.1 INTRODUCTION

This section summarizes the methodology that was followed in carrying out the WFEE project. It is our belief that to be credible and useful, a technical analysis such as the WFEE must: (1) be based on sound technical information and interpretations, (2) follow a process that considers all available data, and (3) incorporate uncertainties into the assessments. A key mechanism for quantifying uncertainties in the WFEE is the use of multiple expert judgments. The *process* used to select the experts, facilitate their interaction and mutual education, and elicit their judgments is just as important as the technical content of their interpretations.

Because of the importance of the entire process, a methodology development team (MDT) was established at the outset. MDT members have experience in developing guidance for and implementing multi-expert studies, understanding technical aspects of waste form degradation and radionuclide mobilization, and performing TSPAs. For example, several members (K. Coppersmith, R. Perman, M. Pendleton, R. Youngs, and P. Morris) have conducted recent expert elicitations for the YMP pertaining to technical issues that include unsaturated zone flow, saturated zone flow and transport, and waste package degradation processes. Drs. Coppersmith and Morris were members of the Senior Seismic Hazard Analysis Committee (SSHAC, described below); and Drs. Coppersmith and Youngs participated on the Electric Power Research Institute (EPRI) methodology team for seismic hazards assessment in the eastern United States (EPRI, 1986). Importantly, the MDT included representatives from performance assessment (W. Halsey and E. Siegmann) and waste package materials characterization (R. Stout and T. Thornton) for the Yucca Mountain project. The design and testing program has developed most of the data related to waste form degradation and radionuclide mobilization for the Yucca Mountain project, and the TSPA will be the primary user of the results of the WFEE project.

2.1.1 Pertinent Guidance Regarding Expert Judgment

In the study of any complex technical problem—such as waste form degradation and radionuclide mobilization at Yucca Mountain—expert judgment is used. The data themselves do not provide an interpretation of the processes or outputs needed for subsequent analyses. For example, experiments such as drip tests or flow through tests do not provide a direct estimate of the intrinsic dissolution models. These data must be interpreted and combined with other data and analyses before they can be used in performance assessments. Through the scientific process, experts integrate and evaluate data to arrive at conclusions that are meaningful to assessments of waste form degradation and radionuclide mobilization, including quantitative and qualitative expressions of the uncertainties. This process is the same regardless of the abundance or scarcity of data; the only difference is the level of uncertainty. In this sense, expert judgment is not a substitute for data; it is the process by which data are evaluated and interpreted. If data are scarce and uncertainties high, the uncertainties expressed by each expert and the range of judgments across multiple experts should reflect that high degree of uncertainty.

The procedures and approaches for eliciting expert judgments, developed through conducting many studies, have been formalized in guidance documents; the process followed in the WFEE project is consistent with these studies. The DOE recently developed guidance for the formal use of expert judgment by the Yucca Mountain Project (U.S. DOE, 1995), and the Nuclear Regulatory Commission (NRC) staff has issued a Branch Technical Position on use of expert elicitation in the high-level waste program (Kotra et al., 1996). Comprehensive guidance on eliciting expert judgment for seismic hazards recently was set forth in a study produced by the Senior Seismic Hazard Analysis Committee (SSHAC) and sponsored by the DOE, the EPRI, and the NRC (SSHAC, 1995).

SSHAC (1995) defines the expert roles of *proponent, evaluator,* and *integrator*. A proponent advocates a particular technical view or interpretation; an evaluator weighs the relative merits of alternative views; and an integrator combines the alternative views into

a composite distribution that includes uncertainties. The WFEE members were informed of their roles as evaluator experts and of the need to forsake the role of a proponent in making their assessments and evaluating uncertainties. Expert interactions are deemed important in the SSHAC process and must be properly facilitated. Finally, the SSHAC process allows for aggregation or combining of multiple expert views. Equal weights are used in the WFEE study to combine the assessments of the experts.

The WFEE study closely follows the procedural guidance set forth in the SSHAC study, both in spirit (e.g., a belief in the importance of facilitated expert interactions) and, in many cases, in details of implementation (e.g., suggestions for conducting elicitation interviews). For example, the WFEE process was designed—in accordance with SSHAC guidance—to result in assessments that represent the "range of technical interpretations that the larger informed technical community would have if they were to conduct the study." However, inasmuch as SSHAC professes to be "non-prescriptive" in specifying a single way of implementing the process, it would be inappropriate to say that the WFEE conforms exactly to the SSHAC process. In some cases, the WFEE process followed approaches that were more appropriate for a relatively modest multi-expert study than a larger, resource-intensive study. For example, after the elicitation interviews, feedback to the experts was accomplished by providing each expert with a feedback package that summarized all of their assessments and the implications of those assessments to certain key issues. The experts then were given an opportunity to revise their assessments in light of the feedback, as suggested in the SSHAC guidance. A more resource-intensive approach might have been to conduct a feedback workshop. Either process enables the experts to review the assessments of others on the panel and to examine the calculated implications of their assessments.

The goal of all of the guidance documents is not to establish a rigid set of rules for eliciting expert judgment; rather, it is to draw from experience—both successes and failures—criteria for when expert judgment should be used and to outline approaches to motivating, eliciting, and documenting expert judgments. Other documents in the

literature provide alternative approaches to the formal or informal use of expert judgment (e.g., Meyer and Booker, 1991).

2.2 WFEE METHODOLOGY

This section of the report summarizes the methodology implemented in the WFEE study. It begins with an overview of the important steps in the process, followed by a more detailed discussion of those steps.

2.2.1 Steps in the Methodology

The principal steps followed in the WFEE project are described below.

1. **Development of Project Plan.** The MDT first developed a project plan that outlined the goals and key elements of the project, timing of significant activities such as workshops, topics to be covered at workshops, and significant milestones.

2. **Selection of the Expert Panel.** The MDT established criteria for participation on the expert panel. These criteria were intended to ensure that all expert panel members had significant professional stature and technical expertise. Highly regarded scientists and engineers were asked for their nominations to the panel, resulting in more than 35 nominations. From this list of candidates, six experts were selected and participated on the panel.

3. **Data Compilation and Dissemination.** The compilation and distribution of pertinent data, including published reference material, began early in the project. The experts were sent a number of data sets and publications throughout the project. Panel members were provided access, if requested, to all Yucca Mountain data gathered as part of the project.

4. **Meetings of the Expert Panel.** Structured, facilitated interaction among the members of the expert panel took place during three workshops. The workshops were designed to identify the significant issues, available data, alternative models, and uncertainties related to waste form degradation and radionuclide mobilization. Debate and technical challenge of alternative interpretations were encouraged to provide that uncertainties were identified. At these meetings, researchers from a

variety of organizations, including LLNL, Pacific Northwest National Laboratory, Center for Nuclear Waste Regulatory Analyses (CNWRA), Nuclear Regulatory Commission, and Atomic Energy of Canada Limited presented pertinent data sets and alternative models and methods.

5. **Elicitation of Experts.** One-day individual elicitation interviews were held with each member of the expert panel. Through discussions facilitated by the elicitation team (all of whom were members of the MDT), each expert provided his assessments of the key issues related to waste form degradation and radionuclide mobilization, expressed the uncertainties, and specified the technical basis for the assessment. The elicitation team documented the elicitation during the interview. The experts subsequently reviewed, revised, and supplemented the summary prepared by the elicitation team.

6. **Feedback of Preliminary Results.** Based on the results of the elicitations, the elicitation summaries from all members of the expert panel were provided to each expert as feedback. This provided each expert with a broader perspective on the range of interpretations being developed.

7. **Finalization of Expert Assessments.** After reviewing the feedback package, the experts developed a final draft of their elicitation summary. The elicitation team reviewed this draft for completeness and clarity, then the experts finalized their summaries for this report (provided in Appendix D).

8. **Preparation of Project Report.** This report was developed to document the process followed, the expert elicitation summaries, and the results.

The rest of Section 2.2 describes in more detail the key activities involved in implementing the WFEE methodology.

2.2.2 Selection of the Expert Panel

The selection of members for the expert panel involved four steps: (1) developing selection criteria, (2) obtaining nominations from knowledgeable individuals, (3) selecting and inviting candidates to participate, and (4) having candidates accept the invitation to participate.

Guidelines for selecting members of the expert panel were developed by the MDT. Candidates for the panel had to demonstrate the following characteristics.

1. Engineer or scientist having a good professional reputation and widely recognized competence based on academic training and relevant experience. Tangible evidence of expertise, such as written documentation of research in refereed journals and reviewed reports, is required.

2. Understanding of the problem area through experience in one or more of the following areas: (a) testing and assessment of long-term chemical and mechanical behavior of inorganic materials (particularly uranium oxides, glass, and Zircaloy) in geologic conditions; (b) testing and modeling of aqueous chemistry and geochemistry of heavy metals and actinides under ambient and slightly elevated temperatures; (c) testing and modeling of transport behaviors of hazardous substances/radionuclides in saturated and unsaturated porous media, including colloidal transport; and (d) testing and modeling of interaction of solids with dissolved species in water for surface complex formation and colloid formation. Experience in the above areas with application to nuclear waste disposal and management is especially valuable. Individuals who have had a major role in the Yucca Mountain Site Characterization Project may be included on the expert panel; however, such experience is not a requirement for participation.

3. Availability and willingness to participate as a named panel member, including a commitment to devoting the necessary time and effort to the project and a willingness to explain and defend technical positions.

4. Personal attributes that include strong communication and interpersonal skills, flexibility and impartiality, and the ability to simplify. Individuals will be asked specifically not to act as representatives of technical positions taken by their organizations, but rather to provide their individual technical interpretations and assessments of uncertainties.

5. Ability to help create a panel of experts representing diverse opinions, areas of technical expertise, and institutional/organizational backgrounds (government agencies, academic institutions, and private industry).

The MDT conducted a broad search to obtain nominations for the expert panel. Letters requesting nominations were mailed to 17 scientists and engineers identified by the MDT. The letters requesting nominations contained a brief description of the project and included the above guidelines for selecting panel members. Thirty-five candidates for the expert panel were nominated and considered in the selection process.

The MDT carefully evaluated each nominee to ascertain whether the selection guidelines had been applied properly, and to balance the panel with respect to knowledge and experience. The list of candidates was narrowed to those individuals who met selection criteria 1 and 2; then the remaining criteria were applied. The professional reputation and publications of each candidate were discussed, and the number of nominations each candidate had received was reviewed. A total of 10 individuals were considered to have the best qualifications for serving as members of the expert panel, and from this group a list of candidates to be invited to participate was developed.

The candidates were contacted by telephone and invited to participate. They were informed that the estimated level of participation was 20 to 25 days. Most accepted during the initial phone call; others requested time to consider potential conflicts of interest or schedule. Six accepted the invitation to become members of the expert panel.

The criteria for selecting the experts were reviewed with each expert before they made a commitment and throughout the project. Each expert had to commit a significant amount of time and would need to prepare for and attend all meetings. Information on potential sources of conflict of interest was provided by each expert and documented in the WFEE administrative files. The experts were informed of the role they would play in the WFEE assessment as an expert *evaluator* who considers a variety of viewpoints, challenges the interpretations of others, and arrives at a reasoned position that includes a representation of the uncertainties. The resulting panel consisted not only of experts of considerable stature and prominence in the professional community, but of individuals with reputations as independent thinkers. The panel members clearly demonstrated that they

were capable of evaluating all the data and hypotheses, and of providing the technical bases for their interpretations and uncertainties.

2.2.3 Review of Technical Issues/Expert Interaction

Technical issues related to the WFEE project were identified in the first workshop and reviewed throughout the project. The workshops provided an opportunity for technical discussion and interaction, with an objective of providing a common understanding of the issues to be assessed and the data available to provide the technical basis for assessment.

Literature and data sets pertinent to assessing waste form degradation and radionuclide mobilization were sent to members of the expert panel throughout the project. Some of the experts made available pertinent publications and reports based on their work at other sites; these publications were made available to the entire panel. A list of the references distributed is provided in Appendix B.

The following sections summarize the workshops conducted during the project. These activities are summarized under the topic of Review of Technical Issues/Expert Interaction, because the workshops were the primary vehicle for accomplishing this interaction. Summaries of the workshops are included in Appendix C.

2.2.3.1 Workshop on Significant Issues and Available Data

The first of the three workshops conducted for the WFEE project was the Workshop on Significant Issues and Available Data. The goals of this workshop were to introduce the panel to the WFEE project; summarize the significant issues related to waste form degradation and radionuclide mobilization process models for the TSPA, and to summarize the data sets developed over the past years related to these significant issues. The workshop included several introductory presentations on topics that included the reference repository design, the waste inventory and the anticipated near-field environment of the repository. Thirteen technical specialists (including members of both the MDT and the expert panel), then presented the data sets and models developed over

the past several years to characterize waste form degradation and radionuclide mobilization at Yucca Mountain.

2.2.3.2 Workshop on Alternative Models and Interpretations

This second workshop conducted for the WFEE project was designed to review the key issues and uncertainties associated with waste form degradation and radionuclide mobilization. Alternative models, modeling results, and interpretations related to the local geochemical and hydrologic environment, cladding degradation, intrinsic waste form dissolution models, secondary phase models, and colloid formation and stability were described and discussed. Thirteen technical specialists made presentations at the workshop.

2.2.3.3 Workshop on Preliminary Interpretations

The Workshop on Preliminary Interpretations, the third and final project workshop, was conducted prior to the elicitation interviews. This workshop provided an opportunity for the experts to present and discuss their preliminary interpretations and uncertainties regarding key issues in waste form degradation and radionuclide mobilization. The experts presented their interpretations of each of six issues: alternative cladding degradation models and rates; waste form surface area exposed; waste form surface in contact with water and water film thickness; intrinsic waste form dissolution modeling; distribution and stability of secondary phases; and mobility and stability of colloids and concentrations.

2.2.4 Elicitation of Experts

Through the elicitation process the experts' interpretations of waste form degradation and radionuclide mobilization at Yucca Mountain were obtained. The elicitations involved a series of activities, which can be grouped into three steps: (1) preparation for the elicitation, (2) the elicitation interview, and (3) documentation and review.

2.2.4.1 Preparation for the Elicitation

Peter A. Morris of the MDT provided elicitation training at the third workshop. The objectives of the training were to demonstrate how to quantify uncertainties using

probabilities, to recognize common cognitive biases and compensate for them, and to present examples of the types of assessments that would be made at the elicitation (e.g., continuous variables, discrete hypotheses, and associated weights). The training was designed to enable the experts to be comfortable with the *process* of eliciting their judgments, so that the elicitation interview could be focused on the *technical issues* of importance to the WFEE.

The specific topics to be covered in the elicitations were the subject of presentations at both Workshops 2 and 3. These topics are discussed in Section 3.0. The list of topics was intended to help the experts prepare for their interviews, to focus their data review on the issues of most significance, and to ensure that the important topics were addressed by all experts.

2.2.4.2 The Elicitation Interview

The elicitations of the expert panel members took place in individual one-day interviews in the San Francisco office of Geomatrix. The interviews were conducted by members of the MDT. The elicitation team was composed of K. Coppersmith (normative expert), R. Stout (specialist), and W. Halsey (specialist).

All data sets provided or made available to the experts during the project were present during the elicitation interviews. The elicitation interview followed a logical sequence from general to more specific assessments and proceeded through all of the technical issues discussed in Section 3.0 of this report. Alternative models, approaches, and hypotheses were discussed and the basis for assessments of uncertainties were also given. The elicitation team took written notes of all assessments during the interviews.

2.2.4.3 Documentation and Review

Documentation of the expert elicitations began with notes taken by the elicitation team during the interviews. Experience on several other expert assessment projects has shown this approach to be preferable to other documentation methods (for example, written questionnaires or experts writing their interpretations following the interview). During

the one-day interview, each WFEE expert was asked to make many assessments, to quantify his uncertainties, and to provide the technical basis for his interpretations. By having the elicitation team take notes, the expert was free to focus on thinking through his answers and thoroughly expressing his interpretations. The elicitation team was able to be flexible in the elicitation sequence (i.e., following the logic comfortable to the expert) while providing that all elements were covered.

Following the interviews, the elicitation team provided each expert with written documentation of the interview, organized by model component. The experts were instructed to review, revise, and expand their preliminary assessments in this "First Draft" documentation summary so that it fully reflected their interpretations. The summaries revised by each expert became the "Second Draft" document. The "Second Draft" summaries from each expert were then distributed to all members of the panel as part of the feedback package so that each expert could review the judgments of others and the technical basis for each judgment.

After reviewing the feedback package, the experts made additional revisions to their elicitation summaries to reflect any resulting changes in their judgments. These revised summaries became the "Third Draft" set. The "Third Draft" summaries were reviewed by MDT members as part of the report review process. The experts responded to any requests for clarifications, and the summaries were finalized. The final elicitation summaries are provided in Appendix D.

2.2.4.4 Feedback and Sensitivity

Feedback to the experts occurred throughout the WFEE project, primarily through interaction among experts. By presenting their ideas on models and interpretations at workshops and in general discussions, the experts both provided and received feedback from their peers on the panel.

More formally, feedback was provided to the experts in several other ways.

- At Workshop 3, the experts presented to the panel their interpretations of key issues. Discussions included the technical bases for the interpretations.
- Written elicitation summaries were provided to all panel members for their review.
- The MDT reviewed the First Draft written elicitation summaries for adequacy and completeness of documentation of the technical bases for judgments.

The feedback-revision process required the experts to defend/revise their assessments and provide appropriate documentation. In all cases, the experts on the panel responded positively to technical criticisms of their interpretations and to reviews of their documentation. The resulting assessments and finalized elicitation summaries reflect the significant effort expended by each member of the expert panel.

2.2.4.5 Aggregation of Expert Assessments

The approach taken to combine, or aggregate, the expert assessments is equal weighting. Importantly, this approach was not a "default" but a goal throughout the project. Accordingly, the proper conditions were created throughout the project to provide that a deliberate, defensible decision could be made to use equal weights (after SSHAC, 1995). The actions that were taken to create these conditions included:

- carefully selecting highly qualified experts who represent diverse views and experience;
- establishing the commitment of each expert to provide the required effort throughout the project;
- identifying a comprehensive data base and disseminating it to all experts;
- educating the experts in issues important to WFEE and training them in elicitation methodologies and the role of experts as evaluators;
- facilitating interaction of the experts in workshops to foster a free exchange of data and interpretations and scientific debate of all hypotheses;

- providing feedback and sensitivity analyses to the experts; and
- providing an opportunity for experts to revise their assessments in light of feedback.

It should be noted that, in accordance with the guidance provided by the SSHAC study (1995), conditions could have been such that differential weights would have been necessary. For example, if a member of the expert panel had been unwilling to forsake the role of a *proponent* who advocates a singular viewpoint, for that of an *evaluator* who is able to consider multiple viewpoints, that expert may have been given less weight or removed from the panel entirely.

3.0 ASSESSMENT OF KEY ISSUES

3.1 INTRODUCTION

The experts involved in the Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) project addressed a variety of technical issues related to the processes and characteristics of degradation of spent fuel and high-level waste glass in the potential repository at Yucca Mountain. The key issues that the WFEE panel members were asked to address are given in Table 3-1. Included are issues important to the Total System Performance Assessment (TSPA); e.g., the potential for taking credit for the delay in fuel dissolution due to cladding, the fraction of the radionuclide inventory released rapidly at the breach of the cladding, the intrinsic dissolution model that describes the rate of fuel dissolution and availability of soluble species). In addition, the experts were asked to provide their perspectives on issues related to conceptual models (for instance, modeling of processes of advective and diffusive transport, models for the mechanisms for development of secondary phases and their solubilities, and the formation and stability of colloids). Finally, the experts were asked for their perspectives on additional data collection or modeling activities that could serve to reduce the uncertainties in the characterization of waste form degradation processes. The experts' evaluations are given in their elicitation summaries in Appendix D and are summarized in Table 3-2 and this section.

It is important to remember the context of the experts' evaluations. First, the goal of this expert elicitation was to characterize the *uncertainties* associated with various aspects of waste form degradation so that the TSPA-VA can incorporate a range of uncertainty when important processes are modeled. As a result, the experts focused considerable attention on what is known from the available Yucca Mountain data and *what is not known*, and the reasons for that lack of knowledge. The reasons could include data gaps, complexities in mechanisms and processes, nonpertinent data, or multiple models consistent with the data. Rather than merely identifying and acknowledging the

uncertainties, the experts were required to provide—to the extent possible—their characterization of the uncertainties for certain key issues, with reference to the technical literature when applicable. For example, the experts provided their assessments of the reactive surface area of the fuel that should be used in dissolution models, utilizing both Yucca Mountain databases that they considered most pertinent and their own experience. Likewise, the experts characterized their knowledge and uncertainty regarding the mechanisms and stability of secondary phases. In light of the sparse data gathered on this subject for the Yucca Mountain project, their assessments express a large degree of uncertainty.

It is also important to note that the experts' evaluations are, to a large extent, an expression of the professional judgment of each expert and are not based on extensive modeling or calculations carried out for this study. These judgments are derived not only from a consideration of Yucca Mountain data, but also data and observations from previous experience. Members of the panel were given a limited time in which to review Yucca Mountain data sets, test results, and models. Further, the experts were unable within the time-frame of the project to conduct their own analyses and calculations. In many cases, the experts relied on the results of analyses presented in workshops by the Yucca Mountain researchers, their colleagues on the panel, or other researchers.

The results of the WFEE expert assessments have potential application in both the upcoming TSPA-VA and the ongoing site characterization program. The TSPA is a probabilistic analysis that is capable of incorporating uncertainties, including those expressed by the WFEE expert panel. The panel members reviewed the modeling of waste form degradation processes carried out for the previous iteration of the TSPA (TSPA-95). By focusing on the issues of most importance to the TSPA, their evaluations will provide useful information on parameter uncertainties and will improve the models to make them more realistic and to better characterize the uncertainties. Likewise, comments made by panel members on the attributes of and suggested improvements to the intrinsic dissolution and solubility models will help improve those models as well. Several members of the panel, who have considerable experience in designing and

conducting laboratory testing programs, provided advice regarding focused experiments that, they believe, could reduce significantly the present level of uncertainty. For example, most members of the panel applauded the flow-through and unsaturated tests conducted to date and encouraged a significant expansion of the experimental effort to include a wider range of fuel types, environmental conditions, and time scales.

3.2 SUMMARY OF EXPERT INTERPRETATIONS

This section summarizes the interpretations made by members of the WFEE panel regarding key issues. The members of the panel expressed a wide range of views on attributes, physical processes, and responses of spent fuel and high level waste (HLW) glass waste forms that may or may not be important to geologic disposal. The intent of this section is to provide the reader with a perspective on the evaluations made by the experts, the manner in which each issue was addressed, an overview of the technical bases for the interpretations, the uncertainties identified by the experts, and the degree of convergence or divergence in the aggregate range of interpretations across the panel. The summary here is not intended to be exhaustive. For a more complete exposition of the interpretations made by the experts, the reader is directed to the elicitation summaries provided in Appendix D.

The key issues listed in Table 3-2 are discussed below.

3.2.1 Overview: Important Waste Form Degradation Processes

A key issue to waste form degradation processes cited by the members of the panel is the environmental conditions that will develop within the waste package, particularly the amount and quality of water entering the package and contacting the fuel or glass. Some experts speculated that early breach of the waste package by a limited number of small penetrations may lead to lower than ambient oxygen partial pressures. However, the uncertainties regarding the potential for this process and its longevity are significant. Important environmental parameters cited include temperature, oxygen availability, water chemistry (pH and [Cl^-]) and the presence or absence of seepage (drips). Some experts

postulated that, because of temperature gradients between the interior and exterior of the waste package, evaporation will occur within the package and lead to concentration of fluids. High Cl^- solutions can have a significant influence on dissolution rates of fuel and glass.

Several experts concluded that fuel cladding—its longevity and mode of failure—are potentially important issues. As discussed below, the experts also concluded that, despite its potential importance to waste degradation processes, there are very few data regarding the long-term behavior and failure mechanisms of the cladding.

Other issues identified as important to waste form degradation include alteration of the fuel and development of secondary phases, the retention of radionuclides in those phases, and the dissolution rates of those secondary phases. Some experts cited the rapid release fraction, including the cladding gap and grain boundaries, as being potentially important to an assessment of peak dose performance.

3.2.2 Cladding

The experts on the WFEE panel were asked for their assessments of the mechanisms of failure of the fuel cladding and the manner in which the long-term behavior of cladding should be modeled. It was generally was agreed that the in-reactor failure mechanisms are well-documented and relatively rare. It is unlikely but possible that juvenile failures of the waste package could lead to aggressive attack of the cladding under high temperatures and oxidizing conditions. Processes of cladding failure under these conditions include creep rupture, stress corrosion cracking, oxidation and delayed hydrogen cracking. More likely, ruptures of the waste package will occur after temperatures decline and relative humidities rise.

Panel members postulated a number of failure mechanisms for the cladding after very long times, many of which depend on environmental conditions. Failure mechanisms identified included general corrosion (which all agreed would occur at very low rates); localized corrosion, including pitting or crevice corrosion; hydrogen embrittlement; and

mechanical failure from collapse of internal structures or other sources. It was emphasized that the potential for localized corrosion is dictated by the presence of water and the possibility of developing concentrated chemistries (especially Cl^-) due to evaporation within the waste package and low pH. Dr. Bates estimated a 10% to 20% probability of localized corrosion, given the attainment of low pH and high [Cl^-]. He postulated that high Cl^- solutions developed from the dissolution of glass canisters could communicate through diffusive flow with other canisters containing spent fuel and lead to localized corrosion. Dr. Bourcier noted that microbiologically influenced corrosion could be very important. Dr. Shoesmith distinguishes between the expected behavior of the cladding in the presence or absence of drips. He concluded that the most likely failure mode without drips is embrittlement; in the presence of drips, crevice corrosion is possible. He assessed the probability of its occurrence to be comparable to the probability of initiating localized corrosion on the C-22 inner barrier of the waste package.

The experts were asked for their assessment of the degree to which cladding "credit" (a delay in the onset of fuel dissolution because of intact cladding) should be included in performance assessment modeling. In general the panel members concluded that presently there is insufficient support for significant cladding credit. Principal reasons for this conclusion include a lack of significant testing data on irradiated Zircaloy, a lack of QA qualified data, difficulties in modeling the variation and time-evolution of environmental conditions, and a lack of a strong mechanistic basis for localized corrosion processes. With the exception of Dr. Shaw, the panel believed that additional data and modeling could be conducted to allow for cladding credit in future performance assessments. Dr. Shaw concluded that cladding behavior is difficult to model and poorly characterized because of large—and likely irreducible—uncertainties in environmental conditions and their temporal and spatial variability.

3.2.3 Rapid Release Fraction

The expert panel members were asked to assess the fraction of the total radionuclide inventory that would be released instantly following the breach of the cladding. In

general, the rapid release fraction was judged to include the inventory both from the gap between the fuel and cladding and from the grain boundaries. Most of the experts endorsed the estimates developed by Johnson and Tait (1997). These are based on a study in which the average fission gas release for boiling water reactor (BWR) fuel was estimated, and gas release was correlated with fission product leaching data from various studies. Mr. Johnson argued that it is reasonable to apply these results to Yucca Mountain because the power history of the rods is similar and the conclusions are based on data from studies of U.S. light-water reactor fuel. These estimates are reasonably consistent with the values used in TSPA-95, but would result in some significant radionuclides (e.g., Tc) being released substantially in the rapid release fraction, while others would be increased. Dr. Apted provided his own estimates of rapid release fractions for certain radionuclides. He concluded that the rapid release fraction is very important—perhaps more so than waste form dissolution—to criterion for the assessment of peak dose performance. He also concluded that I-129 volatility was estimated overly conservatively in the past TSPA, and should be studied to obtain more realistic estimates.

3.2.4 Surface Area Exposed and Water Content

The release or mobilization rate of highly soluble nuclides (i.e., dissolution-limited, not solubility-limited nuclides) will depend on the reactive surface area of the waste form and whether the surface is wetted. The panel members generally agreed that at the first failure of the cladding, all of the surface area of the fuel, including the gap and grain boundaries, will be exposed to oxygen and available for gaseous release and oxidation. Dr. Shoesmith noted that early waste package failure (within the first 100 to 150 years) could lead to preoxidation and consequent significant volume expansion and increase in reactive surface area. However, assuming that temperatures will have declined by the time of waste package and cladding failure, humid-air conditions will prevail. Panel members concluded that at that time essentially all of the surface area of the fuel will be wetted with at least a thin, discontinuous adsorbed water layer. The nature of the cladding failure (e.g, cracks, pits, holes) will to some extent determine the timing of wetting.

Reactive surface areas range from the geometric area of the fuel pellet to the geometric area plus the fragment surface areas plus grain boundaries. Most of the experts agreed with the conclusions of Gray and Wilson (1995) that the reactive surface area to use for waste form dissolution models is the geometric area plus the fragment area plus 4 to 10 grain boundaries (a factor of ~15X to 18X the geometric area). Some experts suggested that the reactive surface area would be expected to evolve with time—both upward and downward—and with environmental exposure conditions. For example, Dr. Shaw concluded that there is good evidence that the relevant surface area will evolve quickly from the surface area of fuel fragments toward that of individual grains. Dr. Apted agreed that there are mechanisms that may lead to increased surface area with time (e.g., mechanical disintegration and oxidation of grain boundaries), but some mechanisms may lead to decreases (e.g., development of alteration layers). Members of the panel noted that models of fuel dissolution and transport should take into account the presence of advective and diffusive pathways and the characteristics of cladding failures. For example, Dr. Bates found it reasonable to include in the models both advective and diffusive pathways, and to take credit for the fraction of the surface area that experiences advective and diffusive flow. Dr. Shoesmith noted that alteration products will form on the surface of the fuel. In the absence of drips leading to advective flow, the alteration products will block fuel corrosion and release; in the presence of advective flow, the protectiveness of such deposits is significantly decreased. Dr. Apted recommended additional work to support the notion that the characteristic failures of the cladding and the container may limit the transport rate of reactants into, and nuclides out of, the pin or container.

Regarding HLW glass, panel members concluded that studies suggest that the increase in surface area due to cracking is about 20 to 30 times the geometric area of the glass log. Dr. Bourcier noted that the development of clays and zeolites could decrease the effective surface area with time.

3.2.5 Intrinsic Dissolution

Intrinsic dissolution, which is the forward reaction rate of the spent fuel (or glass) matrix, defines the mobilization rate for non-solubility controlled elements such as I^{129}, Cs^{135}, Tc^{99}, and colloidally transported elements. (Note that Dr. Bates made a distinction between intrinsic dissolution and forward reaction). The panel reviewed an empirically based regression model developed by the Yucca Mountain project (presentation by W. Halsey, WFEE Workshop 2) in which the dissolution rate is a function of temperature, carbonate concentration, oxygen concentration, pH, and burnup. A series of cross terms describes the interactions between the various parameters determined from a parametric fit.

In general, the experts acknowledged that the intrinsic dissolution model developed by the project is an empirical model based on regression fits to experimental data, rather than a mechanistic model derived from thermodynamics. They concluded, however, that—although a mechanistic understanding of fuel dissolution is desirable—an empirical model is sufficient for purposes of performance assessment. Dr. Bourcier noted that compared to uncertainties in the data, which themselves are relatively few, the differences among various regression models are insignificant. The panel was essentially split on the adequacy of the present model. Three experts considered the present model reasonable and useable in its present form. For example, Dr. Shaw concluded that, with reservations, the dissolution rate model recovers available experimental data to within a known uncertainty, is computationally tractable, depends only on known or predictable environmental and waste-form-related variables, and is valid over the range of variables accessed during the TSPA simulation. These experts also recommend using sensitivity analyses to evaluate the effect of fixing some independent variables.

Three experts recommended that the regression be modified. Dr. Apted and Mr. Johnson suggested that the regression be re-run with temperature as the independent variable. Dr. Apted suggested that burnup also be an independent variable and that the correlation between temperature and burnup be considered. Mr. Johnson recommended that criteria be established for the inclusion of data defined by the range of probable conditions in the

repository. For example, O_2 concentration should be atmospheric; all variations in CO_3 can be included; pH ranges from 7 to 10.5; and burnup can include both UO_2 and spent fuel. After data that meet these criteria are selected, a regression can be run that is a function of temperature only. Several experts also noted that the effect of Ca and Si on fuel dissolution may be significant and should be evaluated given the concentrations expected in Yucca Mountain groundwater.

Dr. Shoesmith suggested that the parameters shown to have a significant influence on fuel corrosion are surface area, temperature, and concentrations of oxygen and carbonate. He suggested an alternative model having the form of a chemical rate equation, which would incorporate the above factors. The intrinsic dissolution rate would be modified with time only by surface area and temperature.

Three experts addressed the issue of modeling the intrinsic dissolution rate of HLW glass. Dr. Bourcier noted that the forward reaction rate of glass depends primarily on pH, SiO_2, and temperature. He further noted that the water entering the waste package is expected to be essentially benign, with pH near neutral and saturated in SiO_2. Dr. Bates concluded that the glass dissolution/reaction rate will vary as a function of water contact mode and the extent of glass hydration. He further concluded that additional data are needed to relate the short-term reaction of glass to long-term behavior. Dr. Shaw provided his view of the shortcomings of the present semi-empirical glass dissolution model developed by the project, including that there is a poor theoretical basis for the long-term rate; the model does not account for dissolved species other than H_4SiO_4 and H^+; and the model may be overly conservative inasmuch as it does not account for precipitation of radionuclide-containing secondary phases.

3.2.6 Stability of Secondary Phases

The intrinsic dissolution rate defines the availability rate for non-solubility limited species. Important issues are formation and stability of secondary-phase minerals that are developed during alteration of the spent fuel or glass, the degree to which these secondary phases retain various radionuclides, and the "effective solubilities" of the secondary

phases that develop. Considering spent fuel first, there was general agreement that a reasonable level of knowledge exists regarding the phases that will form and their sequence under expected repository conditions. However, it also was concluded that, despite test results suggesting retention of radionuclides in secondary phases, sufficient data have not been developed to understand whether the elements of interest reside within the alteration phases or sorbed onto their surfaces. For example, Dr. Shaw recommended that one needs to demonstrate that the radionuclides are in solid solution in the secondary minerals and that the measured partitioning is reversible and represents an equilibrium situation.

Solubilities for various radionuclides ("elemental solubilities") used in TSPA-95 were presented to the panel, as were new results from unsaturated tests that suggest the retention of certain radionuclides in alteration phases. Modeling using the new test results suggest that the "effective solubilities" of the secondary phases are significantly lower (by, in some cases, several orders of magnitude) than the values used in TSPA-95. Panel members were asked whether the new data and modeling provided a sufficient basis for revising the solubilities used in the TSPA. The general conclusion across the panel was that, although the new test results and modeling suggest that the TSPA-95 solubilities are likely conservative, we are not yet in a position to rely on the new effective solubilities in TSPA-VA. For example, Dr. Bourcier concluded that it is premature to use the results in TSPA until it is determined that the processes that control radionuclide concentrations in the thin water film of the small-scale, short-term service condition tests are the same processes that control radionuclide concentrations in the repository. Dr. Shaw concluded that the model used to interpret the results has not been validated, and has been applied only to a small data set at only one drip rate. Dr. Shoesmith noted that the new test results are intriguing but could be invalidated by the uncertainties about whether colloid transport could occur. Taking issue with combining "elemental" and "effective" solubilities, Dr. Apted stated that to rely on effective solubilities, considerations of thermodynamics and predicted phases must be merged with the phases that are actually observed. He and most of the panel members suggested that

the new effective solubilities be used in sensitivity analyses for the TSPA in order to gain an understanding of the potential importance of this issue to repository performance.

Dr. Bates concluded that the new test results are credible but cannot yet be relied on because we have not demonstrated where the elements of interest reside within the alteration phases. He recommended that the effective solubilities for all radionuclides be assessed by weighting the new test results heavily, but incorporating the old range of solubilities from TSPA-95 as a low-probability tail of the distribution.

For use in TSPA-VA, the effective solubility of Np has been revised downward from solubilities used in TSPA-95 (presentation by D. Sassani at WFEE Workshop 2) considering the basis for the previous experimental evidence, which was metastably supersaturated, and evidence that Np is being incorporated in solid solution in secondary phases of spent fuel. The expert panel members were asked for their opinions regarding the proposed revisions presented by D. Sassani. In general, the panel members agreed that the proposed changes appear to be reasonable, although, as noted by Dr. Shaw, experimental data are needed to justify further reductions in estimates of solubilities.

A potentially important issue in modeling the dissolution of spent fuel is the rate that should be used after the UO_2 matrix has been dissolved (i.e., should the intrinsic dissolution model for UO_2 be used to the model the rate?). Those experts addressing the issue concluded that after the UO_2 is completely reacted to hydrous uranyl oxides and silicates, the subsequent release will be controlled by the dissolution rate of those secondary phases. Dr. Bourcier noted that it is important to remember that after the UO_2 is completely altered, the system is closer to equilibrium; the alteration minerals that form are now equilibrated with the solution phase. Therefore, the dissolution rate of secondary phases will be lower than before, and the mobilization rate will be lower than for spent fuel.

In terms of HLW glass, Drs. Apted and Bourcier indicated that evidence exists that Tc, Cs, and Sr are incorporated into glass alteration products such as clay phases and zeolites.

Dr. Bourcier concluded that an average retardation factor can be used to represent ion exchange that retards the transport of radionuclides. None of the experts considered a retardation factor ready for inclusion in the TSPA-VA base case.

3.2.7 Colloids

The panel was asked to consider the mobilization and stability of colloids, but not the transport of colloids out of the waste package or through the near-field environment. There was general consensus across the panel that the spent fuel, and particularly the HLW glass, provides ample opportunities for the formation of colloids. Sources of colloids from spent fuel include, as an upper bound, all of the alteration products and ferric oxyhydroxides from oxidation of the waste package and the internal support structures within the waste package. The panel also concluded that clays developed from altered glass are a likely source of colloids and that tests show that Pu and Am are readily mobilized from glass. Dr. Bates postulates that Pu released from glass in the colloidal state may dominate radionuclide release from the repository. Dr. Bourcier concluded that changes in pH and the ionic strength of pore fluids in the near field will affect the mobility and stability of colloids.

The general conclusion across the panel was that colloids will be readily mobilized by the dissolution and alteration of fuel and glass. Therefore, the assessment of the importance of colloids to performance will be in assessments of how the colloids may floculate, filtrate, or dissolve, which are transport issues and thus not included in the present elicitations.

3.2.8 Recommended Activities to Reduce Uncertainties

Each member of the WFEE panel provided recommendations for activities that can be carried out to reduce uncertainties. These recommendations are given in their elicitation summaries in Appendix D.

TABLE 3-1
ISSUES ADDRESSED BY WFEE EXPERT PANEL

At WFEE Workshop 1, Robert W. Andrews (M&O/DESI) stated that the WFEE expert elicitation would provide information to enhance models and properties/parameters to be used in TSPA-VA. These issues are given below and provided a basis for the elicitations.

MODELS

- Consider range of likely local (e.g., on waste form) geochemical and hydrologic environments
- Evaluate alternative cladding degradation models
- Assess alternate models of waste form surface area exposed and in contact with water
- Define uncertainty and variability in intrinsic waste form dissolution models or alternatives
- Assess alternative models for colloid formation, stability, and mobility

PROPERTIES/PARAMETERS

- Fraction of waste form surface exposed
- Fraction of waste form surface in contact with water for range of local hydrologic environments
- Correlation of water film thickness to local hydrologic conditions
- Intrinsic dissolution rate and correlation to key environmental parameters
- Percent of radionuclides in gaseous and aqueous form
- Distribution and stability of secondary phases (or "effective" solubility limits)
- Mobile stable colloidal phase concentrations

TABLE 3-2
SUMMARY OF KEY ISSUES

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Important Waste Form Degradation Processes	+ The environmental conditions within the package, particularly the quantity and quality of water that enters the package, are important to waste form degradation. + Important environmental conditions include: the temperature-time history and spatial variability; whether drips will occur; the quantity of flow that might occur and whether flow might be episodic or continuous; the chemistry of the incoming water, especially the pH, redox, and concentrations of Cl^-; and the geometry of the failure of the waste package. + Cladding failure is potentially important; any attempt to credibly model failure modes and failure rates should be based on time-dependent, intensive parameters such as temperature and water composition and their effect on Zircaloy. + Once the cladding has failed, release of radionuclides from the gap will be the most important contributor to peak dose for non-sorbing, long-lived, volatile relative fission products.	+ The first information required to perform long-term predictions is an understanding of the environment and identification of materials. + Proper tests address the "most likely" conditions that will be present in the repository, bounding conditions, and conditions that may accelerate the reaction of materials. + In developing a testing program for the unsaturated environment at Yucca Mountain, one recognizes that the repository will be basically dry and that immersion, although possible, is not the expected condition.	+ Water contact scenarios and water chemistry likely will be very important to waste form degradation processes. The scenarios should be coupled with the thermal history of the waste package: whether the water is vapor or drips and, if drips, how much water flows into and through the waste package will be significant to various processes. Oxygen may be driven off early after breach of the waste package. A hot, oxygenated environment could be a strong driving force. Carbonate brines, which develop through evaporation, could dominate the dissolution processes. + A significant process is the interplay between advective flow and diffusive flow, including the fraction of the waste packages and of the waste form that will be involved in aqueous processes.	+ The manner in which waste package failure occurs will affect the environment within the package. Small, local failures of the CRM would affect the inflow of water/water vapor and mass transport. + The potential for failure of the cladding is an important issue if the TSPA includes credit for cladding integrity. + The geometry of the cladding failure can strongly affect fuel dissolution and subsequent transport. + Modeling should consider the potential for radionuclides to be retained in alteration products and episodically released from sudden influxes of water, a scenario that might lead to release of a pulse of radionuclides.	+ Once a waste package has been breached, issues related to the reaction of air (oxygen), water vapor, and liquid water with the waste forms (including the cladding of spent fuel) will be important. + It is important to consider the spectrum of possible environmental scenarios and the evolution of those scenarios temporally and spatially. This is a difficult modeling problem because the waste packages have "memory," and the release behavior of a waste package depends on its entire history up to that point. + An early-breached waste package could, if the penetrations are limited to small openings, have lower-than-ambient oxygen partial pressures within the package. Because of the uncertainties, it will be necessary to assume that the atmosphere in a failed package is the same as the surroundings. + Evaporation of water in the waste package may affect the corrosion rates of metals and possibly the dissolution rate of glass and fuel.	+ The sequence of important events comprises cladding degradation, dissolution of the instant release fraction, waste form dissolution and the release of radionuclides incorporated in the fuel matrix, and the influence of secondary phase formation on fuel dissolution and radionuclide retention. + A variety of failure mechanisms are possible for fuel cladding including those that are independent of repository and those that could occur after failure of the waste package. + The environment within the waste package will be important in determining fuel corrosion rates. In particular, parameters such as temperature, oxygen availability, water chemistry, and the presence or absence of seepage (drips) will be important. + Deposits of corrosion products will form on the fuel surface and may affect radionuclide release by: (1) inducing cladding rupture and increasing surface area; (2) decreasing the release rate by blocking the fuel dissolution process; or (3) immobilizing radionuclides (e.g., ^{237}Np, ^{231}Pa, ^{239}Pu) by coprecipitation.

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Cladding	+ Environmental exposure conditions are important: flux, concentrations of incoming water, characteristics of the cladding, and time-dependence. + Three most likely failure modes are juvenile failures, hydrogen embrittlement, and localized corrosion. + Modeling of cladding failures could lead to a distribution of progressive failures. + It is difficult to take containment credit for cladding because of the need to rely on non-QA data from non-irradiated cladding material.	+ Evaluation of spent fuel cladding should begin with an examination of failure modes. + General corrosion, localized corrosion, hydrogen embrittlement are potential failure modes. + High Cl^- solutions from nearby HLW glass canisters could enhance localized corrosion processes. + TSPA should not take credit for cladding, because tests have not been conducted on irradiated Zircaloy.	+ There are not good models for the modes and rates of degradation of cladding after water is made available. + General and localized corrosion are possible, and microbiologically influenced corrosion is potentially important. Models for the influence of mechanical loads on the cladding do not appear to be well developed. + Because of these uncertainties, it is not at present prudent to take credit for the integrity of cladding. + 304L stainless steel pour canister for glass may become sensitized over long periods; not appropriate to take credit for canister.	+ Pre-container failures include possible creep rupture, stress corrosion cracking, and delayed hydrogen cracking; these are all highly improbable for low to moderate burnup fuel. + For post-container cladding failure, one needs to consider the possible conditions inside the failed canister, including oxygen availability, whether drips occur, temperature and relative humidity, steam, advective flow, perhaps ponding of water within the package, and concentration due to evaporation and buildup of salts in the residual water. + Low probability of localized corrosion, given attainment of low pH and high Cl^-. + Good mechanistic study of localized corrosion needed to make strong argument for cladding credit.	+ Possible failure mechanisms: in-reactor failures (well-documented), failures during transport and loading, localized corrosion (especially with wetting and drying), hydride reorientation, and mechanical failures (e.g., rockfall). + Cladding "credit" can mean distributing failures over time or assuming cladding acts as barrier to diffusive and advective transport; both require the definition of a number of highly uncertain parameters. + The project should try to find some other means of providing an equivalent reduction in the source term, preferably from consideration of an engineered barrier, not a factor as poorly characterized and difficult to model as the cladding.	+ Two basic means of failure: in-reactor creep rupture, corrosion in the respository. + Possible corrosion processes: pitting/stress corrosion cracking, crevice corrosion, general passive corrosion, and embrittlement /hydrogen-induced cracking. + Without drips, most likely process is embrittlement, with stresses from impact from collapse of internal structures or accumulation of fuel alteration products; expect cladding lifetimes of about 10^3 to 10^4 years. + With drips, crevice corrosion is possible, and probability of occurrence is the same as that for localized corrosion on the C-22 inner waste package barrier, with failure as soon as aggressive conditions are established. Failure sites will be small and will impede transport.
Rapid Release Fraction	+ Suggest the following revisions: Cl-36 (10 %); Tc-99 (0.25%); Cs-135, I-129; and Se-79 (3%).	+ Rapid release fraction will consist of the gap inventory and in part the grain boundary inventory; comprises a small fraction of the total inventory of elements such as Np and Tc for fuels examined to date; but could change with higher burnup fuels. + If the fuel reacts at the rates reported for drip, flow-through, or semi-static tests, a rapid release fraction losses much of its meaning because the reported rates are rapid enough to react most of the fuel within a few thousand years, depending on the temperature. + For glass, the rapid release fraction will depend on the extent of vapor hydration that occurs prior to contact with dripping water.	+ May need an analogous process for glass; reservoir of radionuclides from vapor hydration washed away suddenly with pulse of water.	+ Study conducted by Johnson and Tait (1997) estimated the average fission gas release for BWR fuel and correlated gas release with fission product leaching data from various studies. + These estimates are reasonable to use for this project because the power history of the rods is based partly on data from U.S. LWR fuel studies. + Instant release fractions are conservative in that they include both the gap and grain boundary inventories.	+ Results of study by Johnson and Tait (1997) appear to be reasonable. +Fractional fission gas release provides a reasonably defensible upper bound on fractional gap and grain boundary inventory; only a few elements (notably Cs and I) even approach this limit. + Will be necessary to have a defensible projection of the distribution of fission gas releases in both the existing stock of spent fuel and all future fuel destined for a Yucca Mountain repository.	+ Suggest using the instant release fractions developed by Johnson and Tait (1997); these take into account the available data and characterize uncertainties. + There may be some delay in release related to the geometry of failure of the cladding, such as slits from mechanical defects or pits from localized corrosion; but the delay probably is less than the 100 year time steps of the TSPA analyses.

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Surface Area Exposed and Water Contact	+ 100 percent of surface area of the fuel, including the gap and grain boundaries, is available for gaseous release and oxidation. + Fraction of fuel that is wet and available for aqueous dissolution processes ranges from a high of all the grain boundaries (assume a 10 to 30 μm grain size) to a nominal estimate provided by Gray and Wilson (1995) of a factor of 18 times the geometric surface area, or 3.9×10^{-3} m^2/gm. + Uncertainty because some mechanisms may lead to increases in surface area with time (e.g., mechanical disintegration and oxidation of the grain boundaries), while some lead to decreases (e.g., development of alteration layers). + Characteristics of failures of cladding and the container may limit transport rate of reactants into and nuclides out of the pin/container, but additional work is needed to take credit for cladding.	+ At the first failure of the cladding, all of the spent fuel, including grain boundaries, will be exposed to oxygen and oxidation. + In a humid-air environment, all surfaces including grain boundaries, will develop a thin water film, but the reaction that occurs on grain boundaries will not be the same as the reaction on the outer surfaces of fragments. On the outer surfaces, dissolution and alteration is occurring, involving the cations from the groundwater; on the inner surfaces, dissolution/oxidation is occurring + Realistic model of surface area would take into account the exposure conditions and short-term and long-term changes in conditions. + Reasonable to include in the models both advective and diffusive pathways, and to take credit for the fraction of the surface area that experiences advective and diffusive flow. + For glass, the increase in surface area due to cracking is about 20X to 30X within a glass log.	+ For spent fuel, depends on the temperature and oxidation history. + Reasonable surface area for the dissolution model is the geometric area plus the fragment surfaces plus 5 to 10 grain diameters (~15X over geometric area); the development of secondary phases probably will lead to a long-term decrease in this surface area. + For glass, studies suggest 20X to 25X the geometric area of the glass log; development of clays and zeolites could decrease the effective surface area with time. + Under humid-air conditions, vapor hydration will occur and could be followed by episodic advection, leading to a pulsed release.	+ The amount and timing of the exposure of fuel surface area are related to the nature of cladding failures, which could be cracks and/or local corrosion. + The implications to fuel dissolution/alteration of the low cladding-failure surface area are dependent on environmental conditions. + Assuming high relative humidities, essentially all of the fuel surface area will be wetted with at least a thin, discontinuous adsorbed water layer. + Applicable surface area for the matrix dissolution model is 2X to 5X fuel fragment surface area.	+ For a humid-air environment, it should be assumed that all surfaces will be wetted; this should be distinguished from transport due to wetting. + Wetted area is important only to the calculation of the release rate of highly soluble nuclides, including those present in the rapid release inventory. + There appears to be good evidence that the relevant surface area available for contact by water will evolve quickly from the surface area of fuel fragments toward that of individual grains. + Secondary alteration products form on the surface of the fragments and might ultimately fill the inter-fragment void space. At present, we have insufficient data to support the use of fragment area as a conservative case.	+ Early waste package failure could lead to preoxidation and ~150-fold increase in surface area. + Surface area estimated by Gray et al. (1993, 1994) of the geometric area plus the fragment area plus 4 to 10 grain boundaries (a factor of ~15 times the geometric area) is a reasonable surface area to use in waste form dissolution models. + Once T<95°C, and the cladding has failed, all of the surfaces will be wetted. + Alteration products will form on the surface of the fuel. In the absence of drips leading to advective flow, this deposit will block fuel corrosion and release; in the presence of advective flow the protectiveness of such deposits is significantly decreased.

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Intrinsic Dissolution	+ Although desirable, a mechanistic understanding of the processes is not necessary. + Temperature, redox, and carbonate are likely to be most important; Si , Ca, and burnup are likely to be secondary. + Suggest that the regression be conducted with only temperature and burnup as independent parameters and including the correlation between the two. Use a threshold value of Ca and Si in Yucca Mountain pore waters. Exclude the low O_2 data because not relevant to Yucca Mountain. + Run sensitivity analyses and blind tests to see importance of various factors.	+ "Intrinsic dissolution" is the process that occurs in the presence of de-ionized water and will vary as a function of temperature; the rate is an intrinsic property of the material itself. "Forward reaction" rate is dependent on a variety of conditions (pH, ions, burnup, etc.). + The present forward reaction model for spent fuel appears to be reasonable. It is useful to incorporate a range of data on pH, burnup, HCO_3, Ca, Si, etc. May be wise to separate out the dissolution data for UO_2 and spent fuel in the data base. + The glass dissolution/reaction rate will vary as a function of water contact mode and extent of glass hydration. We have a look-up table for the rates of reaction under different conditions, but the data are incomplete. It is likely that the glass will hydrate quickly. + Relate short-term reaction of glass to long-term behavior.	+ All the spent fuel models are regression fits to experimental data, with the form of the equation consistent with a macroscopic, thermodynamically based model. Given the uncertainties in the data, the differences in the regression models will be insignificant. + The present intrinsic dissolution model is reasonable; it does not, however, include the Ca - CO_3 effects examined by Shoesmith et al. (1996). + Retaining a burnup term may be useful for later sensitivity analyses that will allow for radiolysis and fission products to be considered in an empirical way. + Forward reaction rate of glass depends primarily on pH, SiO_2, and temperature; incoming water is essentially benign expected pH is near neutral, and water is saturated in SiO_2. Two measured values for the dissolution rate of the SRL-202 glass under silica-saturated conditions are 0.0016 and 0.0025 $g/m^2/day$. Note that these rates do not include retention of alteration products.	+ Criteria should be established for including data in the pool, defined by the range of probable conditions in the repository. (In other words, O_2 concentration should be atmospheric; all variations in CO_3 ; pH range from 7 to 10.5; and burnup can include both UO_2 and spent fuel.) Then a regression can be developed that is a function of temperature only. + Influence of Ca and Si could be included in modeling fuel dissolution, although it is not apparent that there are sufficient Ca and Si concentrations in the expected water chemistries to achieve a significant effect. + Intrinsic dissolution rate defines the alteration rate, but is an upper bound on the radionuclide availability rate.	+ Intrinsic dissolution model is only important for non-solubility controlled elements, such as Tc, I, and colloidally transported elements. + A fully mechanistic, scientifically rigorous dissolution rate model is not needed. What is needed is a rate expression that: "recovers" available experimental data to within a known uncertainty, is computationally tractable, depends only on known or predictable environmental and waste form-related variables, and is valid over the range of variables accessed during the TSPA simulation. + With reservations, the empirical fuel dissolution rate expression appears to satisfy these requirements. + Suggest that the existing regression fit be used, and that the effect of fixing some independent variables be evaluated by sensitivity analyses. + The glass dissolution model is a semi-empirical model. Its shortcomings include: poor theoretical basis for long-term rate; it does not account for dissolved species other than H_4SiO_4 and H^+; it may be overly conservative inasmuch as it does not account for precipitation of radionuclide-containing secondary phases.	+ Many parameters potentially could influence the corrosion rate of fuel: the degree of surface oxidation, burnup, O_2 concentration, water radiolysis, groundwater composition, pH, and temperature. + Parameters shown to have a significant influence on fuel corrosion are surface area, temperature, and concentrations of oxygen and carbonate. An alternative model would have the form of a chemical rate equation for these parameters: $R_D = k f(A)\, f(T)\, f(GW)\, [CO_3]^p\, [O_2]^q$, which can be reduced to: $R_D = k^1 f(A)\, f(T)\, f(GW)$ i.e., the intrinsic dissolution rate k^1 would be modified with time only by surface area and temperature (f(GW) would not be expected to be time-dependent).

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Stability of Secondary Phases	+ Suggest that drip tests be conducted that titrate the element of interest, look at the concentrations as a function of time, identify intermediate phases, and establish a true solubility limit. + For the TSPA-VA, not reasonable to combine the "elemental" solubilities from TSPA-95 with those "effective" solubilities coming out of the recent low-drip-rate and vapor tests. The two should be kept as separate conceptual models: elemental solubilities versus mobilization rates. + To rely on effective solubilities, considerations of thermodynamics and predicted phases must be merged with the phases that are observed. I suggest that the rates derived from the new data be used in sensitivity analysis, but not be relied on at present. + For rates when all of the UO_2 has dissolved, one needs information on the alteration products and their element content. + For glass, evidence suggests that elements such as Tc and Cs, rather than displaying extremely high solubility limits, are incorporated into glass alteration products such as clay and disordered zeolite.	+ Natural analogs, calculations, and tests with UO_2 all indicate that the same sequence of secondary phases forms. + It is important to evaluate the effects of groundwater on alteration phases and to look at the degree to which various radionuclides are incorporated in the secondary phases. + The new test results are credible because they are based on expected conditions, but it is not yet possible to rely on them completely because we have not demonstrated where the elements of interest reside within the alteration phases. + I recommend that the effective solubilities for all radionuclides be assessed by weighting the new test results heavily, but incorporating the old range of solubilities from TSPA-95 as a low-probability tail of the distribution. + To completely revise the effective solubilities, calculations and experiments must be conducted that demonstrate and explain how the radionuclides are incorporated into or are adsorbed onto alteration phases.	+ Need to determine whether the processes that control radionuclide concentrations in the thin water film of the small-scale, short-term service condition tests are the same processes that control radionuclide concentrations in the repository. It is premature to use the results directly in PA. +The case for changing the distribution of likely Np solubilities to Sassani's recommended values is credible. The new recommended values are consistent with knows mechanisms and with available experimental data. + After the UO_2 is completely altered, the system is closer to equilibrium. The alteration minerals that form are now equilibrated with the solution phase. Therefore, the dissolution rate of secondary phases will be lower than before, and the mobilization rate will be lower than for spent fuel. + For glass, an average retardation factor can be used to represent ion exchange that retards the transport of radionuclides. There is evidence that Cs and Sr exchange into clay phases. A model is needed for the dissolution of the clay alteration products after the glass has been entirely altered.	+ Given voluminous quantities of U(VI) alteration products in the waste package, the range of Np concentrations from spent fuel experiments appears to be directly relevant and should be used. Allowing for a maximum value that is several orders of magnitude higher, as given in Sassani and Siegmann (1998), seems sufficiently conservative. + Pure-phase solubility values may be used in defining bounding value/worst-case calculations. + In modeling the mobilization of solubility-limited radionuclides, it is prudent to consider a scenario in which there is alteration and accumulation of nuclides of alteration products, followed by a pulse of water that washes out the alteration products.	+ From the tests conducted to date, we know the secondary phases that form, but one needs to identify the important secondary phases and show that the radionuclides are in solid solution in the secondary minerals and that the measured partitioning is reversible and represents an equilibrium situation. + Do not recommend that the effective solubilities derived from the low-drip test results be used in TSPA-VA. The model used to interpret the results has not been validated and has been applied only to a small data set at only one drip rate. They could be included in the PA as a sensitivity test to see how important the issue of secondary phase stability is. + After the UO_2 is completely reacted (to hydrous uranyl oxides and silicates), continued release will be controlled by the dissolution rate of these secondary phases. + Revision of Np solubility by Sassani may be reasonable, but strongly recommend that the project try to obtain experimental data to settle questions such as these.	+ Evidence for the blockage of fuel corrosion by the accumulation of corrosion product deposits is extensive, but it is presently impossible to quantify this effect, except for the suppression factor $f(GW)$ of 0.01 (in the intrinsic dissolution model). Therefore, it is judicious to ignore this effect in present modeling efforts. + Regarding the unsaturated spent fuel tests, given the uncertainties about whether colloid transport could occur, I recommend that in the short term (i.e., for TSPA-VA), the source term based on solubilities be retained, and that these observations be used only qualitatively to demonstrate the conservatism of the approach used. If, in the longer term (for TSPA-LA), further evidence in support of these observations can be obtained, then I definitely recommend their use.

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Colloids	+ Relative to colloid formation, stability, and transport, the transport issue is probably the most important. + It is difficult to imagine degradation colloids going far in film transport without advective flow. + Degradation colloids in advective flow conditions are difficult to model because there are no data on agglomeration, chemical dissolution, or filtration. + At most, all of the alteration products are potential colloids, especially in the presence of advection due to dripping.	+ Classify radionuclide-bearing colloids into three categories: pseudocolloids found in groundwater, real colloids incorporating radionuclides in their structure, and primary colloids derived from waste form degradation. + With spent fuel, initially the amount of actinides released as a colloidal fraction is about 50 percent of the total. As secondary phases form, the percent of actinides in colloids decreases as the secondary phases block direct advective flow pathways from the fuel. As more secondary phases form, and their volume becomes greater, they may not be supported as part of the original structure and may spall off the reacted fuel, thus increasing the fraction of actinides released in a colloidal fraction. + Tests to date show that Pu and Am are readily mobilized and transportable from glass, both as a 100 percent colloidal fraction or as 100 percent soluble. The Pu released from glass in the colloidal state may dominate radionuclide release from the repository.	+ Clays developed from alteration of glass are a potential source of colloids. +Although probably only a small portion comes off as colloids, 50 to 90 percent of Pu is mobilized as colloids. Where glass is mixed with spent fuel, actinides released from the spent fuel will sorb onto the colloids generated by the glass alteration. In addition, ferric hydroxide colloids from iron oxide corrosion products will be available for transporting radionuclides from both glass and spent fuel. + Changes in pH and the ionic strength of pore fluids in the near field will affect the mobility and stability of colloids.	+ Tests show that a significant fraction of the Am and Pu is colloidal. + Issues concern how colloids flocculate, the mass transfer that occurs, etc. It is conservative to assume that the colloidal production rate equals the dissolution rate, and then consider the mobilization and transport properties of the colloids.	+ There are abundant sources of colloids, including Pu itself, zeolites, waste package materials, and ferric hydroxides. Glass will also generate clay and zeolite colloids. + Colloids need to be dealt with in transport through the waste package and the near field.	

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
Recommended Activities to Reduce Uncertainties	1. Stop confusing uncertainty (a property of the analyst) with variability (a property of the repository system). 2. Stop relying on future properties of failed engineered barrier materials (container, cladding, waste form) that are assumed to remain favorable throughout the repository lifetime. 3. Include the issue of colloids generated at the waste form surface and migrating into the geosphere in the TSPA. 4. Further development and extension of the oxidative dissolution model for spent fuel ought to be postponed until a rigorous sensitivity analysis of this parameter is made with respect to meeting a dose-rate criterion. 5. A closer connection should be made between solubility studies/thermodynamic estimates of radioelement solubilities and the measured radionuclide concentrations from tests of waste-form alteration. 6. Introducing water that already contains trace (but measurable!) amounts of dissolved radionuclides into existing drip-style tests might promote the growth of alteration products that contain measurable amounts of trace radionuclides. Confidence in this activity would also benefit if an a priori co-precipitation model is used to make blind predictions of reduction factors in concentration for several radioelements between	1. Identify the conditions that will exist as a function of time in the waste package environment: temperature, water availability, air composition, groundwater composition, etc. 2. Identify materials that will be in the waste package and finalize the waste package design. 3. Identify the major modes of reaction (see ASTM 1174-C) for spent fuel and glass corrosion. 4. Identify the key physical parameters that affect glass and spent fuel; decide for which parameters enough information is known for input to LA; and work to get information about those that require it. 5. Agree on submodels for corrosion of glass and spent fuel; convince anyone, someone, that they make sense; demonstrate that they are consistent with the data and have predictive capability; and use them in the TSPA effort. 6. Determine how to account for radionuclide release as affected by formation of secondary phases for both glass and spent fuel. 7. Address key issues related to cladding credit. 8. Address key issues related to corrosion of spent fuel, e.g. effect of radiolysis on corrosion rate, stability of phases as a function of changing disposal conditions, effect of secondary phases on cladding and spent fuel integrity, mechanism of spent fuel reaction, effect of burn-up on radionuclide release.	1. Measurements of thermodynamic solubility are needed, especially related to Neptunium and the kinetics of NpO_2 precipitation. The same tests should be done for PuO_2, for which the same question arises. 2. A need for thermodynamic data exists for U, Np, and Pu silicates. It would be useful to have data on the partitioning of these elements into uranyl silicates in order to estimate the solubility limits for Pu and Np due to these phases. 3. We need more testing of the silica-saturated dissolution rate of real production glassses. 4. Evaporation of J-13 water in a dripping environment will produce a concentrated brine. We need data on spent fuel reacting in such brines. 5. Actinides from dissolving spent fuel could sorb onto colloidal clay particles generated during glass alteration. Corrosion tests should be performed on both glass and spent fuel to evaluate this possibility. In addition, the can-in-can glass canisters, which contain the Pu-ceramic, almost certainly will be a site for this scenario, in which actinides from another waste form contact the colloidal clay particles from DWPF glass. Some tests should be performed to evaluate this scenario. 6. Not much work has been done to couple the materials degradation processes. Additional work should focus on the most potentially harmful coupled interactions. For example, water that has a very high pH due to alteration by cement when it contacts glass will have a much faster reaction rate. Both	1. A good case can be made for cladding credit only if it can be shown that localized corrosion of Zircaloy will not occur under the range of conditions expected within a failed waste package. Because this appears to be the only mechanism that might lead to failures, it would be valuable to perform studies of the initiation of localized corrosion of Zircaloy. 2. The most probable exposure mode for the spent fuel appears to involve a steam environment (adsorbed water film), for which few data are available, and none for temperatures near 100°C. The alteration rate should be measured under such conditions.	1. The dissolution-rate regression has not been tested by attempting to predict the results of tests that were not used in the regression (e.g., other data from AECL or Spain). Doing this would provide a better estimate of the true uncertainties in the predictions. 2. Data are needed on higher/lower burn-up fuels and over a wider range of pH to show whether the putative dependences are real. 3. The errors in the regressed data should be formally propagated (i.e., the data regression should used weighted observations). 4. I suggest that the existing regression fit be used, and that the effect of fixing some independent variables be evaluated by sensitivity analyses. 5. An experimental program should be launched immediately to provide the supporting data for secondary phases. Ideally, one would experimentally determine a water-mineral partition coefficient for each element that does not form its own phase. Such data are needed for each phase and for a range of water chemistries. 6. Additional work is necessary to validate the effective solubilities, such as a new drip test at drip rates between those of the vapor hydration tests and the existing low-drip-rate test. 7. I strongly recommend that the project try to obtain experimental data to settle questions such as Np solubility	1. Composition of the water contacting the cladding and the waste form must be established. 2. The mode of water ingress is required to determine the cladding corrosion function and the ability of deposits to block fuel corrosion. 3. A cladding failure function that incorporates the possibility of localized corrosion should developed. 4. The possibility that deposited secondary phases could block fuel dissolution and retain radionuclides should be investigated. 5. The ability of colloids to retain and transport radionuclides should be investigated. 6. The importance of alpha radiolysis for fuel dissolution remains unclear.

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TABLE 3-2
SUMMARY OF KEY ISSUES, (Continued)

	MICHAEL J. APTED	JOHN K. BATES	WILLIAM BOURCIER	LAWRENCE H. JOHNSON	HENRY F. SHAW	DAVID W. SHOESMITH
	ingressing and egressing water. These predictions can then be compared against the test results. 7. The issue of iodine volatility should be resolved by direct spectroscopic measurement of iodine isotopes obtained by vacuum-pumping the gas volume from freshly breached, cladded fuel rods.	9. Address issues related to integration of EBS components with waste form reaction, both with respect to incoming water composition and transport of radionuclides and colloids through the EBS. 10. Address the information needed about primary colloids (what they are, where the radionuclides are located, their physical properties, and whether they can be transported).	modeling and experimental studies should be used to address this issue.			

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4.0 SUMMARY AND CONCLUSIONS

The Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) project is one of a series of projects that call on the elicitation of experts to characterize the knowledge and uncertainties in key inputs to the Yucca Mountain Total System Performance Assessment (TSPA). The objective of the WFEE project was to characterize the uncertainties surrounding a variety of technical issues related to the processes and characteristics of degradation of spent fuel and high-level waste glass in the potential repository at Yucca Mountain. An understanding of waste form degradation processes is critical to evaluating the performance of the potential high-level nuclear waste repository at Yucca Mountain.

A major goal of the project was to capture the uncertainties involved in assessing waste form degradation processes. So that the analysis included a wide range of perspectives, multiple individual judgments were elicited from members of an expert panel. The panel members, who were experts from within and outside the Yucca Mountain project, represented a range of experience and specialization. A deliberate process was followed, consistent with procedural guidance regarding expert elicitation methodologies, to facilitate interactions among the experts, train them to express their uncertainties, and elicit their interpretations.

The WFEE experts evaluated several key technical issues related to waste form degradation and radionuclide mobilization. A brief summary of the expert interpretations is given below.

Overview: Important Waste Form Degradation Processes

A key issue to waste form degradation processes cited by the members of the panel is the environmental conditions that will develop within the waste package, particularly the amount and quality of water entering the package and contacting the fuel or glass. Important environmental parameters cited include temperature, oxygen availability, water

chemistry (pH and Cl^-), and the presence or absence of seepage (drips). Some experts postulated that, because of temperature gradients between the interior and exterior of the waste package, evaporation will occur within the package and will lead to concentration of fluids. Several experts concluded that fuel cladding—its longevity and mode of failure—are potentially important issues. Other issues identified as important to waste form degradation include alteration of the fuel and development of secondary phases, the retention of radionuclides in these phases, and the dissolution rates of these secondary phases. The rapid release fraction, including the cladding gap and grain boundaries, was cited as important to an assessment of peak dose performance.

Cladding

The expert panel members were asked for their assessments of the mechanisms of failure of the fuel cladding and the manner in which the long-term behavior of cladding should be modeled. Panel members postulated a number of failure mechanisms for the cladding, many of which depend on environmental conditions. Failure mechanisms identified included general corrosion (which all agreed would occur at very low rates); localized corrosion, including pitting or crevice corrosion; hydrogen embrittlement and failure from stresses resulting from collapse of internal structures or other sources; and mechanical failure. The experts were asked for their assessment of the degree to which cladding "credit" (a delay in the onset of fuel dissolution because of intact cladding) should be included in performance assessment modeling. In general the panel members concluded that the degradation of cladding may be important to performance, but there is not sufficient support at present to take cladding credit. Principal reasons for this conclusion included a lack of significant testing data on irradiated Zircaloy, a lack of QA qualified data, difficulties in modeling the variation and time-evolution of environmental conditions, and a lack of a strong mechanistic basis for localized corrosion processes. In sum, the experts cited a lack of knowledge and potential but poorly understood long-term degradation processes.

Rapid Release Fraction

The expert panel members were asked for their assessments of the fraction of the inventory (by radionuclide) that would be released instantly following breach of the cladding. In general, the rapid release fraction was judged to include the inventory from both the gap between the fuel and cladding and from the grain boundaries. Most of the experts endorsed the estimates developed by Johnson and Tait (1997).

Surface Area Exposed and Water Content

The release or mobilization rate of highly soluble nuclides (i.e., dissolution-limited not solubility-limited nuclides) will depend on the reactive surface area of the waste form and whether the surface is wetted. Assuming that temperatures will have declined by the time of waste package and cladding failure, humid-air conditions will prevail. The panel members concluded that at that time essentially all of the surface area of the fuel will be wetted with at least a thin, discontinuous adsorbed water layer. The nature of the cladding failure (e.g., cracks, pits, holes) will to some extent determine the timing of wetting. Most of the experts agreed with the conclusions of Gray and Wilson (1995) that the reactive surface area to use for waste form dissolution models is the geometric area plus the fragment area plus 4 to 10 grain boundaries (a factor of ~15X to 18X the geometric area). Some experts suggested that the reactive surface area would be expected to evolve with time—both upward and downward—and with environmental exposure conditions. Members of the panel noted that models of fuel dissolution and transport should take into account the presence of advective and diffusive pathways and the characteristics of failures of the cladding. For HLW glass, panel members concluded that the increase in surface area due to cracking is about 20 to 30 times the geometric area of the glass log.

Intrinsic Dissolution

Intrinsic dissolution, which is the forward reaction rate of the spent fuel (or glass) matrix, defines the mobilization rate for non-solubility controlled elements such as I^{129}, Cs^{135}, Tc^{99}, and colloidally transported elements. In general, the experts acknowledged that the intrinsic dissolution model developed by the project is an empirical model based on

regression fits to experimental data, rather than a mechanistic model derived from thermodynamics. They concluded, however, that—although a mechanistic understanding of fuel dissolution is desirable—an empirical model is sufficient for purposes of performance assessment. The panel was essentially split on the adequacy of the present model. Three experts concluded that the present model is reasonable for use in its present form. Three experts recommended that the regression be modified with temperature as the independent variable. Several experts also noted that the effect of Ca and Si on fuel dissolution may be significant and should be evaluated given the concentrations expected in Yucca Mountain groundwater. Three experts addressed the issue of modeling the intrinsic dissolution rate of HLW glass, generally endorsing the present dissolution model, with some reservations.

Stability of Secondary Phases

The intrinsic dissolution rate defines the availability rate for non-solubility limited species. Important issues are formation and stability of secondary phase minerals that are developed during alteration of the spent fuel or glass, the degree to which these secondary phases retain various radionuclides, and the "effective solubilities" of the secondary phases that develop. Considering spent fuel first, there was general agreement that a reasonable level of knowledge exists regarding the phases that will form and their sequence under expected repository conditions. However, panel members also concluded that, despite test results suggesting retention of radionuclides in secondary phases, sufficient data have not been developed to understand whether the elements of interest reside within the alteration phases or sorbed onto their surfaces. The general conclusion across the panel was that, although the new test results and modeling suggest that the TSPA-95 solubilities are likely conservative, there are insufficient data to support reliance on the new effective solubilities in TSPA-VA.

The expert panel members were asked for their opinions regarding the proposed revisions to Np solubility presented by D. Sassani (WFEE Workshop 2). In general, the panel members agreed that the proposed changes appear reasonable, although, as noted by Dr. Shaw, experimental data are needed to justify further reductions in estimates of

solubilities. In terms of HLW glass, Drs. Apted and Bourcier indicated that evidence exists that Tc, Cs, and Sr are incorporated into glass alteration products such as clay phases and zeolites.

Colloids

The panel members were asked to consider the mobilization and stability of colloids, but not the transport of colloids out of the waste package or through the near-field environment. There was general consensus that the spent fuel, and particularly the HLW glass, provide ample opportunities for the formation of colloids. They concluded that colloids need to be addressed with transport issues and in assessments of the near-field environment.

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APPENDIX A

BIOGRAPHIES OF EXPERT PANEL MEMBERS

BIOGRAPHIES OF MEMBERS OF THE EXPERT PANEL

Dr. Derek Elsworth is Professor of GeoEnvironmental Engineering at Pennsylvania State University. His research focuses on the mechanical and hydraulic behavior of porous and fractured geologic media with application to poromechanics, penetrometer testing, geothermal reservoir engineering, igneous and volcanic processes, degradation of groundwater by mining activities, and contaminant hydrology. He has a particular interest in improving understanding of the mechanisms of permeability enhancement due to mechanical and thermal effects in fractured rock, and the role these mechanisms play in transport processes. In these areas, he has served as an adviser to the United Nations; the Power Reactor and Nuclear Fuels Development Corporation, Japan; and the U.S. Nuclear Regulatory Commission and its Advisory Committee on Nuclear Waste. The author of more than 100 professional papers, Dr. Elsworth received the Manuel Rocha Medal of the International Society for Rock Mechanics. Dr. Elsworth received a B.S. in Engineering Geology (1979) from Portsmouth Polytechnic, UK, and an M.S. in Engineering Rock Mechanics (1980) from Imperial College of the University of London. He received a Ph.D. in Engineering (1984) from the University of California at Berkeley.

Dr. John E. Gale is Professor of Geological Engineering at Memorial University in St. John's, Newfoundland, and President of Fracflow Consultants Inc., also in St. John's. Dr. Gale's research focuses on the experimental and numerical study of the coupled hydraulic-mechanical response of discrete fractures and the impact that these coupled processes have on both single- and two-phase flow and transport properties of fractured rock masses. He has authored or co-authored more than 150 papers and reports for a variety of conferences and journals and has consulted with a range of industrial and government clients. Dr. Gale completed a B.S. in Geology (1968) from Memorial University of Newfoundland, an M.S. in Hydrogeology (1971) from the University of Western Ontario, and an M.S. and Ph.D. in Engineering Geoscience from the University of California at Berkeley in 1973 and 1975, respectively.

Dr. Roger D. Hart is the director of software services at Itasca Consulting Group, Inc., in Minneapolis, Minnesota. Itasca specializes in geotechnical, mining, and civil engineering consulting and research and in developing computer software. Dr. Hart's duties include training client organizations in the operation of geotechnical software, coordinating code development and modifications, and consulting on civil engineering activities. He has been involved in the development, validation, and application of computer models in geomechanics since 1979. Dr. Hart has performed numerous rock mechanics studies associated with the underground isolation of radioactive waste in tuff, basalt, and salt, and he was the program manager for rock mechanics design reviews of U.S. Department of

Energy studies in connection with underground radioactive waste isolation. Dr. Hart holds a B.S. (1971) and M.S. (1972) in Civil Engineering-Soil Mechanics from Ohio University and a Ph.D. (1981) in Civil Engineering-Rock Mechanics from the University of Minnesota.

Dr. Benjamin Ross is president of Disposal Safety Incorporated, a consulting firm in Washington, D.C., that specializes in analysis of groundwater and soil contamination by radioactive and chemical waste. For the Department of Energy, he developed a coupled model of heat transfer and gas flow at Yucca Mountain that was used to predict the migration of carbon-14 from the potential repository. He currently analyzes near-field thermal effects at Yucca Mountain for the Electric Power Research Institute's Methodology Development Team. Dr. Ross also heads European Analytical Services, Inc., which represents Russian institutes selling technical services and products in the United States. Dr. Ross is a member of the Committee on Buried and Tank Waste of the National Academy of Sciences' Board on Radioactive Waste Management and served on the High Level Waste/Carbon-14 subcommittee of the EPA Science Advisory Board. He received his A.B. in physics from Harvard University and his Ph.D. in physics from the Massachusetts Institute of Technology.

Dr. Robert S. Schechter is Professor Emeritus of Chemical and Petroleum Engineering at the University of Texas at Austin. He has chaired the Departments of both Petroleum and Chemical Engineering. After beginning his tenure at the University of Texas in 1957, he held various Professorships and Chairs. He has been a visiting professor at several universities, including Kansas, Edinburgh, and Brussels. In 1976 Dr. Schechter was elected to the National Academy of Engineering; he was named Chevalier of the Order of Palmes Academiques in France in 1980. He has published 215 papers and 5 books. He has received a number of awards for his research, including the American Society of Engineering Educators Senior Research Award (1987), the Franklin Carll Award from the Society of Petroleum Engineers (1994), and the AIME Mineral Industry Faculty Award (1998). He holds a B.A. from Texas A&M University (1950) and a Ph.D. from the University of Minnesota (1957), both in Chemical Engineering.

Dr. Yanis Yortsos is Professor of Chemical Engineering and Petroleum Engineering at the University of Southern California, where he has also served as Chairman of the Chemical Engineering Department from 1991 to 1997. Since January 1995, he has held the Chester F. Dolley chair in Petroleum Engineering. His research and teaching interests are in the general areas of fluid flow, transport, and reaction processes in porous media. Dr. Yortsos has received several honors and awards, including the Rossiter W. Raymond Award of the AIME for Outstanding Technical Paper by an author younger than 33 years

of age (1985) and the Distinguished Service Award of the Pi Epsilon Tau (1984). He currently serves in the Editorial Boards of the journal *Transport in Porous Media* and of the *Society of Petroleum Engineers Journal*. He has been an invited or visiting professor or researcher at institutions that include the Universite Paris in Orsay, the Universite Pierre et Marie Curie in Paris, Stanford University, and Clarkson University. Dr. Yortsos has published more than 90 refereed publications, has given more than 60 invited presentations, and presented more than 90 conference papers. He received a B.S. degree (1973) from the National Technical University, Athens, and M.S. and Ph.D. degrees from the California Institute of Technology (1974 and 1979, respectively), all in Chemical Engineering.

APPENDIX B

REFERENCES DISTRIBUTED TO EXPERT PANEL MEMBERS

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APPENDIX C

WORKSHOP SUMMARIES

SUMMARY
WORKSHOP ON SIGNIFICANT ISSUES AND AVAILABLE DATA

WASTE FORM DEGRADATION AND RADIONUCLIDE MOBILIZATION EXPERT ELICITATION PROJECT

November 18-19, 1997
Clarion Hotel - San Francisco Airport
Millbrae, California

The Workshop on Significant Issues and Available Data was the first of three workshops to be conducted for the Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) for the Yucca Mountain project. The purposes of this workshop were to (1) introduce members of the expert panel to the goals and context of the WFEE project; (2) summarize the significant issues related to modeling waste form degradation and radionuclide mobilization for the total system performance assessment (TSPA) for the proposed high-level nuclear waste repository; and (3) summarize data related to those significant issues. Copies of the handouts and overhead transparencies from the workshop can be found in the administrative files for the project.

DAY 1 - TUESDAY, NOVEMBER 18

Workshop participants were welcomed by Paul Daniel of the Yucca Mountain Site Characterization Office. Kevin Coppersmith of Geomatrix then introduced the workshop, identifying the project's Methodology Development Team, describing the purpose and procedure for performing an expert elicitation, and summarizing how panel members were nominated and selected. He described the process and guidelines for the elicitations, the role of the experts, what will be covered in the workshops, and the meetings of the expert panel. Members of the expert panel then introduced themselves: Michael (Mick)

Apted of QuantiSci, who studied source term models for the Yucca Mountain total system performance assessment conducted by the Electric Power Research Institute (EPRI); John Bates of Argonne National Laboratory (M&O/ANL), who has worked with the high-level waste program since 1977; William Bourcier of the Lawrence Livermore National Laboratory (M&O/LLNL), who has specialized in high level waste glass forms and thermodynamics; Lawrence Johnson of Atomic Energy of Canada (AECL), who has specialized in spent fuel and engineered barriers for performance assessments in Canada; Henry Shaw of M&O/LLNL, who until 1988 worked with the Yucca Mountain project on spent fuel and glass waste; and David Shoesmith of AECL, who has specialized in the modeling and electrochemistry of fuel dissolution, the chemistry of uranium oxide, and containers for nuclear waste.

David Stahl, who works on the Yucca Mountain site characterization project through Framatome Cogema Fuels (M&O/FCF), gave the first of four introductory presentations designed to set the stage for specific data and discussions on waste form degradation. He presented the reference repository design, summarizing the concept, location, and layout of the repository and waste packages; emplacement patterns and spacing for packages; and design of waste containers and engineered barrier systems. Stahl also described the inventory of DOE spent nuclear fuel, discussing the characteristics of DOE spent fuel with regard to repository disposal. He also noted where various categories of waste will come from and their current and projected amounts, along with projections of waste from commercial nuclear enterprises.

Robert Andrews of Duke Engineering and Services (M&O/DESI) continued the introductory talks by describing the role of the WFEE in supporting model abstractions for the total system performance assessment-viability assessment (TSPA-VA), which is due to Congress by September 30, 1998. He stressed the importance of the experts' roles as evaluators of waste form performance models, not members of a peer review panel. Andrews described how the key components in the TSPA interconnect and how information moves within a TSPA. He described available models of radionuclide mobilization, the issues surrounding each model, and the information about the barrier

systems those models require. He also provided references for distribution to the experts, along with comments from the Nuclear Regulatory Commission's TSPA peer review panel. He outlined the key issues to be addressed in the expert elicitations, from the geochemical and hydrologic environments to cladding degradation and alternative models for colloid formation, stability, and mobility.

After a short break, workshop participants reconvened to hear William Halsey, M&O/LLNL, , describe issues and alternative models related to waste form dissolution and radionuclide mobilization. He outlined the importance of waste form degradation, the waste forms themselves, and processes that alter the wastes. He described experiments performed and models developed to understand waste form degradation. Halsey reviewed the way that TSPA-95 represented and ranked the issues, and how the model was evaluated at an abstraction workshop in February 1997. He outlined the issues the TSPA-VA must consider and the alternative approaches that could taken to factors such as waste form degradation and the near-field environment. The general discussion of the role of the WFEE in the TSPA was completed by Christine Stockman, M&O/Sandia National Laboratory, speaking for Dave Sassani, M&O/DESI. Her topic was the importance of the near-field environment and how it and its associated uncertainties are defined in the TSPA. She defined and described the near field, emphasizing the native geochemical processes and the effects of heating and of the materials themselves. She then described proposals for testing factors such as microbial effects, gas composition, and reactions between water and solids of interest (cement, steel, and backfill). Stockman showed several schematics for the evolution of environmental actors such as changes to temperature, dissolved oxygen, dissolved chloride, and pH, and discussed the associated uncertainties and possible variations among drifts. After her presentation, the workshop adjourned for lunch.

After the lunch break, the panel members heard presentations on the dissolution of spent nuclear fuel (SNF), which controls the rate of release of radionuclides. The first, by Ray Stout of M&O/LLNL, addressed concepts for a model of SNF degradation and dissolution. He reviewed the inventory of waste forms (primarily spent fuel), container

design and degradation, the characteristics of the SNF, and the responses of the wastes in the repository (oxidation, dissolution, alteration, and release). His group has developed a generalized model of dissolution (forward reaction rate), which is being used as the basis for modeling the growth of layers of altered material, because the rate of dissolution establishes the rate of alteration. He also presented the burnup dependence for the dissolution of LWR SNF, then reviewed information acquired about the formation of alteration layers. Stout concluded that release rates are limited by solubilities and alteration layers and depend on the types of water flow (drip, vapor phase, etc.). He also concluded that if water flow is localized, vapor flow will be the primary mode of dissolution, alteration, and release.

Next, panel members heard three descriptions of laboratory tests on SNF. Next, panel members heard three discussions of laboratory tests on SNF. The first, by Walter Gray of Pacific Northwest National Laboratory (PNNL), was titled, "Flowthrough Dissolution Testing of LWR Spent Fuel." He described the apparatus and conditions for 20 tests performed to date. After giving the equation for intrinsic dissolution used in TSPA-95, he noted that it was based on results of only the first nine tests. Gray then applied intrinsic dissolution rates to the wetted surface of fuel in a repository. He showed dissolution rates and estimated depth of water penetration into grain layers for unoxidized and oxidized spent fuel. He compared results from flowthrough and drip tests for ATM-103 fuel in J-13 water at 90°C and outlined plans for future tests. Steven Steward, M&O/LLNL gave the next presentation on dissolution of SNF. His group studied the rate of dissolution in "aggressive" groundwater. Their flow-through experiments varied oxygen, pH, carbonate, temperature, and fuel burnup. Steward described the different modeling approaches investigated and his preferred model, which allows for multivariate modeling. Results matched well the results of experiments with carbonate and oxygen levels. Dissolution rates increased for higher oxides and at higher carbonate and temperature levels. Results also matched extrapolated UO_2 data well.

After a short break, John Bates, a member of the expert panel, described experiments on unsaturated spent fuel under oxidizing conditions. He and Patricia Finn performed petri

dish experiments designed to measure properties that are not themselves responses to the repository environment. He described the methods and set ups for various types of tests. Results indicated that under high-drip conditions, complete release of radioelements is possible; under low-drip conditions, complete release is inhibited until the drip becomes vapor, at which time complete release is possible. Bob Finch, also of ANL, continued the presentation by describing scanning and analytical (SEM and AEM) electron microscopy performed to identify the products of alteration of SNF. The SEM analysis is designed to identify corrosion and alteration products, estimate the extent of reaction, and infer reaction mechanisms. These tests also were performed under high-drip, low-drip, and vapor conditions.

Ananda Wijesinghe, M&O/LLNL, gave the last talk of the day and the last on SNF release under unsaturated conditions. His group's tests were designed to obtain model parameters for calculating the rate of release of radionuclides (known as the source term). He summarized release rates under unsaturated conditions based on results from the tests Bates and Finn performed for no-drip, low-drip, and high-drip conditions on two spent fuels. Wijesinghe outlined preliminary model assumptions about hydrology and reactive transport and described development of the preliminary conceptual model. He then presented results of data analysis to date and directions for improving data analysis. Both the model and its preliminary conclusions concerning transport of condensed vapor water were validated by results from the experiments of Bates and Finn.

After this talk, observers were given an opportunity to make comments or ask questions. Questions concerned the type of water chemistry, the near-field environment, and the rate of release of plutonium. After presenters responded to questions, the workshop adjourned for the day.

DAY 2 - WEDNESDAY, NOVEMBER 19, 1997

The second day began with a presentation on cladding by Eric Siegmann, M&O/DESI. He described the characteristics of cladding, gave an overview of mechanisms for degradation of cladding both before and after emplacement, and outlined the model of cladding for the TSPA-VA. He discussed cladding corrosion and various mechanisms of corrosion (the prime one being surface oxidation); strain failures; cladding unzipping (from the oxidation of UO_2 to U_3O_8); mechanical failure (e.g., drift degradation); delayed hydride cracking (to date believed to be insignificant in conditions of dry storage); stress crack corrosion (also believed to have little or no role in cladding degradation at the proposed repository). The model of cladding for the TSPA-VA is designed to estimate cladding failure and the resultant fuel surface area and oxidation state of exposed fuel. Conclusions to date include: (1) Limiting temperatures to less than 350° limits the number of strain perforations; (2) Delaying package failures to after 4000 years limits the amount of cladding unzipping and oxidation of UO_2 to U_3O_8, decreasing both the amount of exposed fuel and the dissolution rate; and (3) Using Zircaloy for the container material will reduce the amount of UO_2 fuel exposed to the waste package environment.

Continuing the discussion of cladding, Robert Einziger of PNNL gave a presentation titled, "Fuel Oxidation and its Effect on Fuel Rod Behavior in a Repository." He discussed how oxidation affects release rate (source term), outlining how oxidation of the fuel inside a slightly defected or punctured cladding (causing a phase transformation, a splitting of the cladding, and release of fission gas) could contribute to further cladding rupture (or unzipping). His group has performed various tests to examine fuel oxidation. He described the materials tested, apparatus used, and results for various fuels and their transition products at various temperatures. Splitting of cladding, he said, results when the formation of U_3O_8 stresses the cladding, and the stress allows crack propagation. Einziger's conclusions included: (1) Although the transition of UO_2 to U_4O_9 generally is well understood, the transition of U_4O_9 to U_3O_8 is poorly understood; (2) Excessive fuel oxidation will severely degrade the fuel into a powder, rapidly (depending on

temperature) splitting the cladding open to expose the fuel; and (3) Experiments give no indication that cladding would be breached under conditions of dry storage.

Next, Cynthia Palmer, M&O/LLNL, gave a status report on the development of a thermodynamic database for geochemical modeling. Defining hers as a "chemist's view" of the proposed repository, she outlined the strategy for updating the database that will support the TSPA-VA. She described the base-case near-field geochemical environment for TSPA-VA, including in-drift materials and locations of different waters. Developing a thermodynamic database begins, she noted, with identifying the types of radionuclides in the anticipated materials and categorizing those elements according to whether additional thermodynamic data are required. "The geochemical model is only as good as the database," she stated, and the database may be inadequate, resulting in a reliance on wide probability distributions around a mean value. The database is important, she said, because solubility, which depends on thermodynamics, has a major effect on release rates and amounts for TSPA-VA. Data sets have been developed for the radioelements of interest and will be used to calculate stable-phase solubility for a given fluid composition; log normal distributions will be constructed for each calculated value based on variability in fluid composition, uncertainties in the data, and uncertainty in the conceptual model.

After a short break, participants reconvened to hear John Bates' second presentation of the workshop, titled, "Colloid Investigations." He described the colloid state and his group's studies of radionuclide-bearing colloids, their growth, stability, and diffusion and sorption properties. Their objective is to find ways to study colloids formed from the reaction of glass or spent fuel with groundwater under unsaturated conditions to determine size fractions, radionuclide content, and physical properties. Predicting the release of radionuclides through colloid transport is important because the generation and migration of colloids from waste form degradation may greatly increase the amount of radioactivity transported from the waste site. Bates described the methods his group used (such as filtration, AEM, and audioradiography) to characterize and measure colloids. He showed results for static and drip tests on glass, and tests on SNF.

Next William Bourcier, M&O/LLNL, also a member of the expert panel, discussed concepts for modeling degradation of defense high-level waste glass. He outlined the mechanisms for dissolution of glass as it reacts with water and forms alteration layers. He described the apparatus and method for a flow-through experiment and compared experimental results with results from model simulations. The tests varied glass composition and solution (for pH, for instance) to study the effects of various elements and of changes in solution. He then outlined the model of glass corrosion proposed for the TSPA-VA, the simplifications incorporated in it, the methodology behind it, and its limitations.

After a break for lunch, Lee Bendixsen of Idaho National Engineering and Environmental Laboratories' National Spent Nuclear Fuel Program, outlined the inventory and characteristics of DOE-EM spent nuclear fuel. He described the distribution of SNF throughout DOE sites by type and amount in terms of mass, volume, mass of uranium, mass of fissile material, and heavy metal. He described the DOE spent nuclear fuel inventory, identifying seven categories and 15 groups of nuclear fuels in their inventory. He identified the various forms the fuel takes, from plates or wafers to rods, pellets, or blankets, and described the cladding, which ranges from Zircaloy-2 and zirconium to stainless steel. Bendixsen also provided the dimensions and geometries of the fuel forms, conditions of the materials, where and how they are stored, and their potential impacts at the repository (e.g., potential carbide-water impacts).

Next Walter Gray of PNNL continued his discussion, "Flowthrough Dissolution Testing of Hanford N-Reactor Spent Fuel." He distributed preliminary data (with disclaimer) from measurements of the rate of dissolution of fuel under conditions relevant to those of the proposed repository to compare it with the rate of dissolution of LWR fuel. His group performed some tests with fuel exposed to a lot of water, conditions unlike those expected at the proposed repository, but comparable to the LWR spent fuel tests, and the corrosion rates were high. Per surface area, the rates of dissolution of uncorroded specimens for N-reactor fuel were higher than for LWR fuel; but on the basis of total inventory in the repository, the LWR and Hanford fuels provided comparable uranium

dissolution source terms. Stage 2 corrosion rates were much higher than Stage 1 dissolution rates. He identified possible mechanisms of Stage 2 corrosion: galvanic or anoxic corrosion (the anoxic mode being much faster).

After a short break, the workshop opened to a discussion among members of the expert panel and presenters. Observers also brought up questions or comments. Then Kevin Coppersmith talked about the next workshop, which he said will be based on the last two pages of the handout from Bob Andrews, concerning areas that the expert elicitation should focus on to enhance the TSPA-VA.

John Bates gave his third presentation, and the final one of the workshop, standing in for William Ebert of Waste Management Department at ANL. The presentation was titled, "Overview of Glass Testing Program to Support Waste Acceptance." The program, which includes vapor tests and 10-year drip tests, is intended to help develop a mechanistic understanding of glass corrosion to determine a source term for PA calculations. The source term includes the reaction rate of the glass and rates of radionuclide release and distribution. Various tests (such as vapor hydration and drip tests) and analyses were performed. Among Ebert's key findings: the alteration layer is not protective; clay colloids form early, then settle out; and release rates for some glass increase after some phases form.

Kevin Coppersmith thanked expert panel members, presenters, and observers, then adjourned the workshop.

SUMMARY
WORKSHOP ON ALTERNATIVE MODELS AND INTERPRETATIONS

WASTE FORM DEGRADATION AND RADIONUCLIDE MOBILIZATION EXPERT ELICITATION PROJECT

December 15-16, 1997
Embassy Suites
South San Francisco, California

The Workshop on Alternative Models and Interpretations was the second of three workshops to be conducted for the Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) for the Yucca Mountain project. The project is sponsored by the U.S. Department of Energy (DOE) and managed by Geomatrix Consultants. The purposes of this workshop were to (1) review the key issues and uncertainties associated with waste form degradation; (2) present and discuss alternative models, modeling results, and interpretations related to the local geochemical and hydrologic environment, cladding degradation, intrinsic waste form dissolution models, secondary phase models, and colloid formation and stability; and (3) identify any additional data needed to conduct the elicitations. Copies of the handouts and overhead transparencies from the workshop can be found in the administrative files for the project.

DAY 1 - MONDAY, DECEMBER 15

Kevin Coppersmith (Geomatrix Consultants) introduced the workshop, identifying its purposes and the role of the experts as evaluators of numerous interpretations rather than proponents of any particular interpretation. He outlined the steps in and schedule for the rest of the project, including the feedback step and documentation of expert assessments. Then Eric Siegmann of Duke Engineering and Services (DES) gave the first of two

overview presentations of the status and purposes of the WFEE. He reviewed the role of the expert elicitations in enhancing models to be used in the total system performance assessment-viability assessment (TSPA-VA). He outlined the properties and parameters to be examined in the TSPA and considered in the elicitations, including alternative models for cladding degradation; intrinsic waste form dissolution rate for light water reactor (LWR) spent fuel and high level waste (HLW) glass and its correlation to key environmental parameters and time, and colloid formation, stability, and mobility; and the correlation of the thickness of water films to local hydrologic conditions. Other factors include the percent of radionuclides in gaseous and aqueous forms, the fraction of the waste form surface that is exposed, and the fraction that comes in contact with water under a range of local hydrologic environments.

In the second overview presentation, William Halsey of Lawrence Livermore National Laboratory (LLNL) reviewed options for modeling waste form dissolution for the TSPA. The goal, he said, is to model the rate of mobilization of radionuclides from the waste forms as a function of time and environmental parameters. There are limited experimental data on spent fuels, and the previous TSPA (TSPA-95) used a combination of bounding dissolution rates and solubility limits that provided for a rather conservative release rate. Halsey outlined the models for predicting dissolution rates for LWR spent nuclear fuel (SNF). He described criticisms of the SNF model in TSPA-95 and results of a model abstraction workshop held to develop a plan for refining waste form modeling for TSPA-VA. He described a revised model for the LWR SNF dissolution rate, which added an oxygen and burnup dependence to the TSPA-95 model and the aspects of aqueous film concentration being examined. Halsey then presented a proposed "release factor" model that 1) incorporates the revised LWR dissolution rate model and 2) provides a thin film aqueous radionuclide concentration model which combines dissolved and transportable particulate species and which could thereby provide a more realistic effective solubility than current solubility ranges for such key radionuclides as plutonium, uranium, americium, and neptunium.

Describing more specific aspects of the TSPA, David Sassani (DES) gave a presentation titled, "Constraints on Solubility-Limited Neptunium Concentrations for Use in Performance Assessments." He presented values derived from both dissolution tests and calculations. Based on a synthesis of dissolution studies, Sassani recommends decreasing the expected value for Neptunium in J-13 type water by about two orders of magnitude. He described the way in which values can be estimated and uncertainties in both conceptual models and model parameters can be evaluated for TSPA. The workshop took a short break after Sassani's presentation.

After the break, Jerry McNeish (DES) gave the first of two presentations on environmental conditions in the waste packages. He discussed the movement and action of water on the engineered barrier system (EBS) at the proposed repository. Key to the movement of water and performance of the EBS are the rate and location of water flow into the drift, onto the waste packages, into the packages, and onto the waste form. Wetting of the waste forms, of course, initiates release, which can be represented by advective, diffusive, or "bathtub" models. McNeish described the TSPA-VA model for water movement in the EBS, a preliminary drift seepage model, assumptions regarding waste package degradation, and waste form wetting scenario(s). He described how the computer code RIP is used to evaluate the EBS and highlighted uncertainties in the EBS flow model, including the percent of waste packages that are dripped on, locations and timing of waste package breaches, and the fraction of waste that becomes wet because of advective flow. Workshop participants stopped for lunch after McNeish's presentation.

After lunch, Kevin Coppersmith briefly described assessments of drip properties with respect to waste packages, drip area, pit size, and so forth made by an expert panel convened for the Waste Package Degradation Expert Elicitation project. Then David Sassani, giving his second talk of the workshop, continued the discussion of environmental conditions in the waste packages by describing what is known of the near-field (in-drift) geochemistry. He defined the geochemical near field and its processes, presenting the underlying assumptions and a base-case conceptual overview. He described the base-case approach to modeling gas and water compositions, the abstraction

of thermohydrologic results, and the strategy for calculating water composition. The forthcoming TSPA will be the first to incorporate changes in water composition into the modeling. Sassani presented results of geochemical calculations based on the composition of the incoming water and the concrete liner. He also mentioned associated uncertainties/variables, such as distribution of salt crusts, the extent of microbial activity and other sources of CO_2, and additional introduced substances.

Cladding degradation was the subject of the next two presentations. The first presentation, by D.G. Franklin of Bettis Atomic Power laboratory, was titled, "The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Environment." Zircaloy, he said, has existed for only about 40 years, but its corrosion is well studied and fundamentally understood. There is a large database of test results to support Zircaloy corrosion rates. Water chemistry and precipitates are the key to understanding the corrosion of Zircaloy, he noted. Experiments have shown that the microstructure of Zircaloy is affected by radiation, so that primary issues are the effect of prior irradiation and effect of the environment, along with the difficulty of extrapolating conditions and effects over time. The accelerating parameter for corrosion of Zircaloy is temperature. Franklin outlined the priority data needs regarding Zircaloy, such as data for long testing periods, low temperatures, and thick oxides. Various experiments have been performed on Zircaloy (including his autoclave tests under high and low temperatures); results indicate that oxide thickness is a significant factor, as water chemistry also may be. He found that there is no indication of change in the mechanisms that control the rate of corrosion as temperatures decrease. Franklin concluded that the general corrosion of Zircaloy would be expected to be less than 1 mil in 1000 years.

After a brief break, panel members returned to hear Eric Siegmann's second presentation of the workshop and the second on cladding: "Cladding Credit in TSPA-VA." The purpose of the model of cladding in TSPA , he said, for each bin of waste packages, is to (1) estimate cladding perforation, (2) estimate fuel surface area exposed through unzipping, and (3) estimate oxidation state (fraction that is U_3O_8). Two types of analyses have been performed: early strain failures and late mechanical failures. For early strain

failures, a statistical model calculates the fraction of pins (of 5,200 sampled) that perforate and unzip. He described the bases and assumptions for strain failure analysis, providing error factors and their weightings. He showed results for commercial PWR fuel. His conclusions included: (1) limiting cladding temperatures to below 350° limits the number of strain perforations; and (2) delaying waste package failures to after 200 years eliminates cladding unzipping and UO_2 oxidation to U_3O_8, decreasing both the amount of exposed fuel and the dissolution rate. The mechanical failure he considered derives from blocks of rubble falling on waste package assemblies.

Next, Lawrence Johnson of Atomic Energy of Canada Limited (AECL) gave the first of six talks on modeling waste form dissolution. His presentation was titled, "Release of Segregated Radionuclides from Spent Fuel." His group is working to develop estimates of instant release fractions (IRFs) for key radionuclides for performance assessments for the Swedish nuclear program (SBK-TR97-18). They are estimating average fission gas release for boiling water reactor (BWR) fuel and correlating that gas release with data on fission product leaching from various studies. Fission gas release (FGR) is given as a percentage plotted against burnup. The best average estimated FGR is 2%; a pessimistic estimate is 4%. Johnson provided IRF data for numerous radionuclides, including Cesium (Cs-137) and Iodine (I-129), which show similar IRFs; CL-36 (which had a higher IRF, perhaps due to a smaller atom size that may cause it to diffuse more rapidly); and Technicium (Tc-99).

The last presentation of the day was given by Yueting Chen, Duke Engineering and Services, Inc. (CRWMS/M&O), who described work his group is doing to investigate in-situ dissolution of spent fuel and radionuclide mobilization. They are using the TSPA-95 dissolution model and aqueous equilibrium correlations for the key species along with the reactive-transport simulator program AREST-CT. Among the program capabilities are that it can consider both equilibrium and kinetic reactions and can couple transport with reactions. He described the mathematical model behind the program and its output in the form of dissolution/precipitation rates, water chemistry, release fluxes of radionuclides and other aqueous species, changes in porosity, and so forth. Chen detailed the two-step

strategy behind the effort: to estimate the dissolution of UO_2 and its secondary phases, then describe interactions among individual radionuclides and the container. He walked through the steps, for Step 1 detailing assumptions and chemical reactions. He established the physical configuration for Step 2, then outlined various reactions, focusing on the concentration of Tc. On-going work includes examining TcO_2, examining the temperature dependency of reactions, and evaluating other radionuclides.

After this last presentation of the day, observers offered comments, including one concerning the importance of whether the water in the model is a silicate or carbonate water. After some discussion, Coppersmith adjourned the workshop for the day.

DAY 2 - TUESDAY, DECEMBER 16, 1997

The second day of the workshop continued the discussion of modeling waste form dissolution, beginning with a presentation by David Shoesmith (AECL.) His presentation was titled, "Fuel Dissolution under Oxidizing Conditions and Models to Predict Fuel Dissolution Behavior." Fuel dissolution, he said, initially should be rapid and maintained, then eventually become insignificant. Shoesmith noted the primary sources of oxidants within a waste vault, then described the evolution of the fuel chemistry under waste vault conditions and corrosion under oxidizing conditions (which produce a corrosion reaction). He demonstrated that the fuel dissolution process is modified by the precipitation of secondary phases, because precipitated layers act as insulators and barriers to transport. Shoesmith went on to describe the redox chemistry of fuel oxidation and dissolution and the choice of an appropriate model. He also described production of radical oxidants and their reaction with UO_2 fuel, which appears to be controlled by diffusion. Aggressive oxidizing conditions, he noted, appear to produce positive corrosion potentials and perhaps the establishment of local acidity, the extent and effect of which are not well defined. Shoesmith next described various models for the corrosion and dissolution of fuel and their applicability to different situations. He showed the dissolution rates different models predict, noting predicted dissolution rates due to alpha radiolysis. The model appropriate to Yucca Mountain conditions, he said, requires a

kinetic equation for fuel dissolution and mass balance equations for various factors. He ended by outlining some of the questions that remain regarding models for fuel dissolution, including the appropriate kinetic equation for fuel dissolution, the effect of any build-up of corrosion products, and whether redox scavenging will occur in the presence of corrosion product deposits.

John Tait (AECL) continued the discussion of modeling waste form dissolution rates. He described dissolution rates both of uranium from UO_2 and uranium and radionuclides from spent CANDU fuel. His group performed work to validate the model of spent fuel dissolution using data from flow-through leaching experiments. The dissolution rate of UO_2 and spent fuel is dependent on dissolved oxygen, temperature, and composition of groundwater (bicarbonate). The rate of uranium dissolution from UO_2 is also dependent on gamma dose from the spent fuel itself. His group compared measured results with predicted results for various materials and solutions. They found that for an aerated carbonate solution, measured rates and orders of reactions matched those predicted by electrochemical models. Agreement was not as good for neutral, non-complexing solutions, for which predicted values were as much as two orders of magnitude less than measured values.

Next, Tae M. Ahn of the Nuclear Regulatory Commission's (NRC's) Office of Nuclear Material Safety and Safeguards (ONMSS) discussed important considerations in modeling waste form source term. Considerations involve the observed dissolution rate of the primary phase of bare spent fuel and mechanisms that potentially could limit radionuclide mobilization, such as the wet fraction of spent fuel in the repository, for which almost no experimental data are available. Other considerations that might limit radionuclide mobilization are cladding protection modes and modes of cladding failure, which include localized corrosion, gamma radiolysis, hydrogen effects, and splitting by expansion of the matrix volume. Protection by secondary minerals, on the other hand, can decrease mass transport and exposed surface area and decrease solubility limits by radionuclide fixation. After Ahn's presentation, workshop participants took a short break.

After the break, Richard Codell of the NRC's ONMSS gave a presentation titled, "Alternative Models of Fuel Wetting and Radionuclide Release in the NRC Total System Performance Assessment, Phase 3." He described model abstractions for isothermal water influx to the drifts, water ingress to and egress from the packages, fuel wetting, and radionuclide transport from the packages. He described the models, their assumptions, and how model results for various parameters are used in TSPA, noting that work is ongoing and the models subject to change. Models for fuel wetting include the so-called bathtub model and a flow-through model, along with two models that are yet to be developed: the pulled-plug model and diffusion-only model. Codell outlined model assumptions and compared release rates associated with the models, noting that preliminary results from TSPA 3.1 indicate that the distribution of peak doses calculated using alternative models for waste wetting are surprisingly similar. NRC staff, Codell said, consider the bathtub model reasonable but are exploring other models of containment and wetting.

The next two presentations concerned analogs to alteration of nuclear waste forms at Yucca Mountain. The first presentation, by David Wronkiewicz of the University of Missouri-Rolla, described natural analogs. The role of natural analogs, he said, is to validate the reaction pathways observed in experiments, identify secondary phases and phase paragenesis, and identify pathways of trace constituents as surrogates for radionuclides. The purpose is to examine the reaction of UO_2 and simulated waste glasses in the unsaturated environment, identify reaction mechanisms for waste form corrosion, identify paragenesis of alteration phases, identify potential alteration phase hosts for long-lived radionuclides, and demonstrate the method for testing UO_2 before testing spent fuel. Wronkiewicz described the conditions and apparatus for a dynamic drip test on UO_2 designed to simulate unsaturated conditions. He found a rapid uranium release in the first one to two years, after which particles became entrapped in a mat of alteration phases. Wronkiewicz went on to describe a vapor hydration test, intended to simulate unsaturated conditions, performed on waste glass. From these texts he concluded again that migration of radionuclides from glass waste might be retarded by their incorporation in alteration phases. After this presentation, workshop participants stopped for lunch.

After the lunch break, panel members reconvened to hear the second presentation on natural analogs, which was the final presentation of the workshop. William Murphy of the Center for Nuclear Waste Regulatory Analysis/ Southwest Research Institute examined the Nopal I mine in the Pena Blanca District of Mexico as a geologic analog for Yucca Mountain. After acknowledging his colleagues working in the area, Murphy noted the advantages of natural analog studies. Only geologic systems, he said, permit direct study of chemical isolation and transport phenomena over the time and space scales required for evaluating a geologic repository. He described the Nopal I uranium mine at Pena Blanca and the features that make the site similar to Yucca Mountain, including that both sites are located in Tertiary volcanic rocks and fractured tuffs. After outlining the uranium minerals identified at the Nopal I deposit, he described the paragenesis of these minerals and then compared the sequence of mineralogic alteration of uranium according to laboratory data from Wronkiewicz and others (1992) with that based on field data from Pena Blanca. He described the series processes that affect the source term, for instance water flow, oxidative dissolution of spent fuel, and dissolution of secondary products. He derived a natural analog release rate for the potential repository by first calculating the maximum average oxidation rate of uraninite at Pena Blanca; this oxidation rate can be used to set a limit on release rate for calculations regarding repository performance. Murphy considers the Nopal I uranium deposit an uncanny analog of the proposed repository at Yucca Mountain. The alteration sequence of uraninite at Pena Blanca resembles the alteration of UO_2 in experiments designed to mimic conditions at the proposed repository. His group is in the process of scaling results from Pena Blanca to Yucca Mountain.

Kevin Coppersmith ended the workshop by outlining future plans and schedules for the WFEE project. He reviewed the purpose of Workshop 3, which will be a forum for expert panel members to present and consider their preliminary interpretations, receiving feedback from other panel members. After the elicitations, additional opportunities will be provided for review of assessments and documentation of the process. There were no comments from observers, so Coppersmith thanked expert panel members, speakers, and observers, then adjourned the workshop.

SUMMARY
WORKSHOP ON PRELIMINARY INTERPRETATIONS

WASTE FORM DEGRADATION AND RADIONUCLIDE MOBILIZATION EXPERT ELICITATION PROJECT

January 27-28, 1998
Embassy Suites
South San Francisco, California

The Workshop on Preliminary Interpretations was the last of three workshops to be conducted for the Waste Form Degradation and Radionuclide Mobilization Expert Elicitation (WFEE) for the Yucca Mountain project. The project is sponsored by the U.S. Department of Energy (DOE) and managed by Geomatrix Consultants. The purposes of this workshop were (1) to provide an opportunity for the experts to present and discuss their preliminary interpretations and uncertainties regarding issues key to waste form degradation and radionuclide mobilization; (2) to provide the experts with feedback from their colleagues on the panel regarding their preliminary interpretations before the elicitation sessions; and (3) to provide the experts with elicitation training. For each of six key issues, two experts presented their preliminary interpretations, after which all of the experts discussed the issue. The focus was on understanding the interpretations, their technical bases, consistency with data, and expression of uncertainty. Copies of the handouts and overhead transparencies from the workshop can be found in the administrative files for the project.

DAY 1 - TUESDAY, JANUARY 27

Kevin Coppersmith (Geomatrix Consultants) introduced the workshop, identifying its purposes and ground rules, which are intended to make the experts the focus of all

workshop activities. He noted that all data and discussion to be presented at this workshop are preliminary and subject to examination and revision. He emphasized that the experts are expected to examine their own and others' interpretations with an open mind and willingness to consider all bases and assumptions. Coppersmith then outlined the activities and schedule for the rest of the project, including the elicitations and the feedback, review, and documentation steps that follow them.

Next began the discussion of the six assigned issues. The first issue to be addressed by the experts was alternative models and rates for cladding degradation, and first to give his preliminary interpretation was Lawrence Johnson [Atomic Energy of Canada Ltd. (AECL)]. Johnson reviewed the modes of degradation for Zircaloy. Degradation modes before container failure are creep rupture, stress corrosion cracking (SCC), and hydrogen-induced cracking (HIC). Degradation modes after container failure are corrosion (general or local) and secondary failure mechanisms. Johnson noted data on creep rupture from dry storage monitoring that indicate low strain failures (< 1% strain) can occur; however, this occurs on cladding with very thick oxide coatings, expected only on very high burnup spent fuels. This would indicate very low probabilities for this failure mechanism at temperatures up to 350°C. Johnson also considers SCC a low probability. HIC [or delayed hydrogen cracking (DHC)] he considers unlikely unless hydrides precipitate in highly stressed regions of the cladding under conditions of slow cooling. The possibility needs to be better addressed, he said; the dry storage data may help resolve this issue. He expressed concern regarding secondary failure mechanisms, which he said he would discuss more thoroughly in his second talk. He found little likelihood of failure from general corrosion, but after a waste package fails, some potential for water to enter the package and evaporate, leaving a salt deposit and the potential for pitting or crevice corrosion. Tests are thus needed on local corrosion by dilute salts at temperatures less than 100°C, and mechanistic studies of cladding pitting and crevice corrosion. Cladding unzipping from oxidation to U_3O_8 appears to be relevant only to containers that fail prematurely where some rods have failed. He suggested an activation energy for the UO2 to U3O8 transition of 146 ± 10 kcal/mole be used in any unzipping analysis. Hydrogen embrittlement is possible, he believes, after cladding failure.

The second interpretation regarding models and rates for cladding degradation was given by David Shoesmith (AECL). He outlined a cladding failure scenario in which cladding either fails directly or remains intact but fails because of corrosion. In either case, fuel exposure and dissolution are the ultimate outcomes. Shoesmith looked at possible corrosion processes on Zirconium and Zircaloys. Pitting and SCC have been observed in sea water, and crevice corrosion, although unlikely, has been reported. General passive corrosion should be very slow (as summarized by Franklin at WFEE Workshop 2). Embrittlement and HIC of Zirconium and Zircaloys require further study, Shoesmith noted. The pitting potential of Ti is well known; pits can develop under only limited conditions, one of which is a saline environment. Zircaloys and Zr-Nb alloys also could pit if the environment became sufficiently saline. Shoesmith examined passive corrosion scenarios for Zr and Ti whereby the materials absorb hydrogen. However, Shoesmith noted, the cladding already contains hydrogen in excess of its solubility. Failure could occur when HIC becomes possible and does occur when gross embrittlement becomes possible. He gave the critical hydrogen limits for embrittlement of Ti and Zr, which he considers the most likely failure process. Given the necessary aggressive environment, Shoesmith predicts the time required for this failure to be 10^3 to 10^4 years. During the discussion that followed, it was noted that the concentration of chloride is important. After the discussion of this first issue, participants took a short break.

After the break, participants returned to hear preliminary interpretations regarding the second issue, the surface area of waste forms that might be exposed. Mick Apted of QuantiSci, Inc., was the first expert to speak. After reviewing the way the 1995 total system performance assessment (TSPA-95) dealt with this issue, the properties assessed and conceptual models proposed, he offered new terms for inclusion into the TSPA-VA. He suggested that the estimates of exposed geometric surface area and wetted fraction of fuel were overly conservative, and the estimates of the enhancement of effective surface area, such as grain boundary exposure, were not sufficiently conservative. Apted defined f_{exp} as the fraction of surface area exposed, then further defined "exposed," outlined the characteristics of failures, and noted uncertainties that affect the application of f_{exp}. He

also presented alternative considerations regarding f_{exp}, such as the fraction of waste packages that fail over time.

Giving his second presentation of the day, Larry Johnson provided the second preliminary interpretation of the amount of waste form surface area that is exposed. He examined the nature of cladding failures, the estimated surface areas of cladding failures, and the significance of exposed surface area when estimating fuel oxidation and release. Cracking failures, he said, result in very small failed surface areas; the area produced by pitting or crevice corrosion is difficult to estimate, but also small compared to the rod surface area. He outlined the implications of a small surface area of failure of the fuel cladding for fuel dissolution/alteration under dry air, saturated air, and steam conditions. He noted that oxidation in dry air conditions was localized near the cladding defect, whereas for saturated air and steam additional oxidation extended to the ends of the rods, well away from the cladding defect. Johnson then showed photographs of oxidation fronts near defects in cladding and reviewed some of the literature, including data on U release from bare fuel, hole defects, or slit defects and data on Cs-137 release. Johnson then outlined some implications of O_2 and U mass balance considerations, for instance the large increase in volume. The discussion that followed Johnson's presentation concerned pitting penetration, exposed surface area, and what issues will be most important to the performance assessment. Workshop participants stopped for lunch after the discussion.

After lunch, participants returned to hear preliminary interpretations on the third issue of concern: the waste form surface in contact with water and the thickness of the water film. The first interpretation was given by Henry Shaw (LLNL). He began by addressing the question, "Why do we care about surface area?' In some cases, he said, we don't, as for solubility-limited releases. The key, he said, is to define concentrations in the liquid phase. Surface area has importance only for elements having concentrations in the fluid phase that cannot be expressed in terms of the concentration in a coexisting solid (in other words, for dissolution-rate limited releases). Shaw went on to examine reasonable scenarios. A fully saturated repository is unlikely, he noted, believing the most likely

cases would involve failed packages under conditions of dripping water or of humid air or steam. After summarizing the data presented/available and the data gaps faced, he concluded that the entire surface area of failed rods must be considered to be wetted during the time scale of interest. He ended by addressing the question, "What is the appropriate particle size for estimating fuel surface area?" He again noted the available data, interpretation, and remaining questions. He considers the relevant scale to be that of fuel fragments, not individual grains.

Mick Apted gave his second preliminary interpretation of the workshop, and the second to examine the issue of waste form surface in contact with water and the thickness of the water film. He discussed the fraction of UO_2 matrix surface area in contact with water given various models ("drips-on" model, bathtub model, and so forth). He also looked at characteristics of the water film, its thickness and flow, referring to some published references. Apted summarized issues and uncertainties surrounding general repository conditions, the fraction of waste form surface area contacted by water (f_{wet}), and the water film. He ended by expressing concern that the derivation of "concentration boundary conditions" in TSPAs to date has been based on speculative, possibly non-conservative source term models. He requested alternative source term models that rely on more readily obtainable data, properties of the engineered barrier system, and simplified boundary conditions. After a discussion with observers, which focused on the thickness of water films that may form on fuels, workshop participants took a short break.

After the break, David Shoesmith gave his second presentation of the workshop, providing his preliminary interpretation of the fourth issue, the modeling of intrinsic waste form dissolution. He described three modeling procedures: an empirical parametric model, a chemical rate equation, and an electrochemical mixed potential model. Intrinsic dissolution (corrosion) rates are based on single-pass flow-through experiments; these rates have been reproducible from lab to lab and are consistent with electrochemically determined values. The temperature dependence predicted by flow-through test results may not be relevant to repository conditions. Shoesmith reviewed the effects of groundwater constituents, then summarized environmental parameters at the repository.

He reviewed intrinsic dissolution rate models, including a model appropriate if the rate is considered to be independent of environmental conditions, and a model having significant measured dependencies. Shoesmith noted that he considers the probability of a drip falling directly on a cladding rupture site to be very low. Shoesmith summarized the most significant parameters controlling dissolution rate: available water, cladding failures, temperature, and accumulation of corrosion products. The intrinsic dissolution rate, he concluded, is best represented by a chemical rate equation based on measured parameters; the drip scenario should be treated separately.

The second preliminary interpretation of modeling intrinsic waste form dissolution, and the last presentation of the day, was given by Henry Shaw (LLNL). In his second talk of the day, Shaw examined the requirements for modeling waste form dissolution for the TSPA viability assessment (TSPA-VA). He then showed a calculated regression of residuals of Yucca Mountain project fuel dissolution rate and noted that other experimental data show a 10x to 100x variability of dissolution rate for different, but nominally the same, sample types and test conditions. He believes that these variables may dominate uncertainty. As with surface area, he said, dissolution rate is important only for elements not controlled by solubility. Shaw then expressed his reservations regarding the regression, among them that the expression clearly is not valid over the range of possible conditions and that it does not account for all possible solution species. As regards the model of glass dissolution, he said that Bill Bourcier would cover the shortcomings of this semi-empirical model. Shortcomings include its weak theoretical basis for the long-term rate and the fact that it does not consider secondary phases.

The discussion that followed Shaw's presentation focused on how results of the expert elicitations will fit into the base case being developed for the TSPA-VA. After some discussion, Coppersmith adjourned the workshop for the day.

DAY 2 - WEDNESDAY, JANUARY 28, 1998

The second day of the workshop provided preliminary interpretations of two more key issues, the first being the distribution and stability of secondary phases. Bill Bourcier (LLNL) gave the first interpretation. He began by listing the key questions surrounding the issue, such as: "Which phases form? Do they incorporate radionuclides? If they do, will the change affect radionuclide release? If so, how?" He also noted that the individual release submodels for glass and spent fuel currently in the TSPA are not coupled, that is, the effect of the release from one component does not adjust the chemistry of the water contacting another component. After noting the data and information available to answer these questions, Bourcier went on to identify additional data needs, such as better data for radionuclide partitioning into alteration minerals and thermodynamic data for all alteration minerals that provide solubility limits for radionuclides in TSPA. He suggested that tests be done in pre-reacted water, tests of glass and spent fuel together, and spent fuel tests in high (>10) pH. He also identified overall TSPA needs, such as better data on the compositions of incoming water and paths of water flow through the repository.

John Bates (ANL) gave the second interpretation of the distribution and stability of secondary phases. Bates focused on glass waste, which he says represents 30 to 50 percent of the waste form and will dominate the solution chemistry. He identified uranium, plutonium, and americium as the only nuclides of concern. Bates presented photographs showing alteration phases, indicating that they do form under vapor and drip conditions. He described the development of stability diagrams, which illustrate the predominant solid or dissolved species in equilibrium with a fluid as a function of two variables. He then provided the parameters for calculating the diagrams and showed the results. Bates also discussed reaction progress plots, which illustrate the expected variations in equilibrium phase assemblages as increments of UO_2 are dissolved in the fluid. The reaction progress plots agree well with test data. Comparing the model predictions with test results, Bates concluded that thermodynamic data for U minerals at 25°C probably are approximately correct, but that solubility data for most U minerals

remain only qualitative. After this presentation (any discussion?), workshop participants took a short break.

Participants reconvened to hear two preliminary interpretations on the mobility, stability, and concentrations of colloids. John Bates returned to give the first interpretation of this issue, giving a brief review of his previous discussion of colloids at WFEE Workshop 2 and defining a colloid as any particle of size from 1nm to 5μm suspended in water. He named the characteristics of colloids, such as diffusion properties, stability, and sorption properties, then described radionuclide-bearing colloids, stressing their importance to the radionuclide-related source term. The generation and migration of colloids from waste form degradation, he said, may greatly increase the radioactivity transported to the site boundary. Colloids will be released from the waste packages, he said. Considerable data are available on colloid formation from glass and on sorption of plutonium by colloids. As regards geochemical modeling, he concluded that we must use source term test data as a starting point. He then noted needed information: better knowledge of radionuclide-bearing phases and the interaction of colloids with materials of the engineered barrier system.

The final preliminary interpretation of the day, and the second on the issue of mobility, stability, and concentrations of colloids, was given by Bill Bourcier. In his second talk of the workshop, Bourcier looked at key questions, available data, and data needs regarding this issue, emphasizing the need for characterizing the generation of clay colloids from glass. Data needs include developing a better idea of the abundance and composition of natural colloids and of what mineral phases exist as colloids. He reviewed the approach taken in the TSPA-VA, the input data, and additional data needed. After brief comments, Bourcier turned the floor over to Christine Stockman (Sandia National Laboratory) for an update on TSPA-VA. She said that the project has added "waste form colloids in containers" and "waste form colloids on glass" as factors to be evaluated. The computer code RIP and data from Inez Triay's group at Sandia are currently being used for analyses. A general discussion of colloid data and the use of the RIP code, in which both members of the expert panel and workshop observers participated, followed.

Coppersmith, expert panel members, and members of the methodology development team (MDT) then discussed the format of the upcoming elicitation interviews. After this discussion, the workshop adjourned for lunch.

After lunch, expert panel members received the elicitation training. Kevin Coppersmith introduced Peter Morris of Applied Decision Analysis, Inc., a member of the project's MDT and specialist in decision analysis, who provided elicitation training. Morris spent the afternoon discussing using probabilities to quantify uncertainty, representing and manipulating probabilities, and assessing probabilities. Probability, he said, is a formal quantification of uncertainty, and probabilities can be assessed either objectively (if the uncertainty pertains to a property of the physical world) or subjectively (if the uncertainty is a degree of belief about the real world). A probability assessment can be viewed as a qualitative assessment of a person's knowledge. Morris reviewed the terminology and calculations used in probabilistic analysis, defining probability density functions, cumulative probability distributions, and expected values. He went on to discuss quantifying uncertainty with probability and using probabilities to support decision-making. Morris also described the motivational and cognitive biases that can creep into probability assessments. He then described the six-phase process of probability assessment and the technique of aggregating expert assessments. After Morris's training, Coppersmith thanked all workshop participants and observers and adjourned the workshop.

APPENDIX D

ELICITATION SUMMARIES

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ELICITATION SUMMARY
MICHAEL J. APTED

OVERVIEW OF WASTE FORM DEGRADATION PROCESSES

Throughout my assessment, my recommendations and conclusions are based on the assumption that the dose-standard set forth by the U.S. National Academy of Sciences Panel (NAS, 1995) will be adopted. In particular, I assume that the safety standard will be based on evaluating peak dose rate whenever it occurs, out to one million years.

Once the waste package is breached, the environmental conditions within the package, particularly the quantity and quality of water that enters the package, are important to waste form degradation. The temperature-time history seems reasonably well estimated, although spatial variability arising from rock inhomogenity and repository layout should be considered. It is also important to know whether drips will occur, the quantity of flow that might occur, and whether flow might be episodic or continuous. For example, the mobilization of colloids might be affected by this behavior, as well as by the mode of water contact with the spent fuel. The chemistry of the incoming water is important, especially the pH and redox. In addition, the concentrations of Cl^- could be important to assessing the potential for corrosion of the cladding. The geometry of the failure of the waste package also could be important. Failing at the top of the package but not at the bottom could have implications to the egress of radionuclides. A "bathtub" may develop, particularly if reaction products clog possible locations of egress. The sensitivity of the "bathtub" and "pulled plug" model should be evaluated as scenarios in the TSPA. It would be a mistake to assume favorable location, timing, and dimensions of localized failure in the waste container and cladding without comprehensive and compelling evidence.

Cladding failure is potentially important; any attempt to credibly model failure modes and failure rates should be based on time-dependent, intensive parameters such as temperature and water composition and their effect on Zircaloy. We should not rely on notions such as physical protection from a precipitate layer, because uncertainties surround the adherence and sustained coverage of such layers. At temperatures > 96°C there will only be adsorped monolayers of water on the surfaces within the waste package. At temperatures <96°C, the heat output could still be high enough that a continuous liquid water phase may not be present on surfaces. Heat output versus time should be compared with the expected rates of water infiltration to assess the potential for evaporation. This calculation allows a credible bounding estimate based on readily measured properties. There may be an extensive period between boiling and the presence of a continuous layer of liquid water.

Once the cladding has failed, release of radionuclides from the gap will be the most important contributor to peak dose for non-sorbing, long-lived, volatile relative fission products. In TSPA-95, Iodine was assumed to be volatile, which had a major adverse effect on its rate of near-field mobilization and transport. Other experiments do not show Iodine emitted as a gas; it is commonly 1/1000th of the gap inventory. I suggest that Iodine and its volatility be examined.

CLADDING

Estimates of the behavior of cladding must consider the environmental exposure conditions that the cladding will be exposed to. The flux and concentrations of incoming water, the characteristics of the cladding, and the time-dependence are all important. The three most likely failure modes are juvenile failures, hydrogen embrittlement, and localized corrosion. General corrosion is not an important process. Juvenile failures at high temperatures can lead to unzipping of the cladding. Unzipping is likely to be rare, based on industry data, but how "rare" must be assessed with greater confidence. Localized corrosion is possible if Cl^- concentrations can be developed, as for example from local refluxing. There appear to be insufficient data on unirradiated Zircaloy to

provide estimates of the rate of localized corrosion or the number and geometry of pits. Long-term hydrogen embrittlement can put the cladding in a condition poised for mechanical failure. The potential for sudden mechanical loads arising from rockfall, earthquakes, collapse of internal supports, and so on should be evaluated by bounding analyses to explore the potential for common mode failure of cladding.

The modeling of cladding failures could lead to a distribution of progressive failures, in the same way that a distribution of waste package failures could be developed. However, it is important to consider possible common-cause failures that could lead to more pulsed releases. It is difficult to take containment credit for cladding because of the need to rely on non-QA data on non-irradiated cladding material. Indeed, the need to take such credit does not seem well-founded. Extending mean time of containment to 10,000 years, or even 500,000 years, does not appear to significantly attenuate peak release rates (or dose rates) of key nuclides for a repository at Yucca Mountain. A possibly more fruitful option would be to look at the possible breadth of the distribution of cladding failures across the repository and examine how sensitive the TSPA results are to this.

SURFACE AREA EXPOSED AND WATER CONTACT

Upon failure of the cladding, the exposure of the fuel to air and water can be considered. By "exposure" I mean the formation of an open, continuous system to the geology and atmosphere outside the engineered barrier system. Therefore, at first failure of the cladding, 100 percent of the surface area of the fuel, including the gap and grain boundaries, is available for gaseous release and oxidation. Tests at Argonne National Laboratory (ANL) and Pacific Northwest National Laboratory (PNNL) show evidence of relatively rapid (~10s of years) oxidation of grain boundaries (Einziger et al., 1992; Thomas and Einziger, 1992; Wronkiewicz et al, 1996).

The continuity and thickness of water films are a function of temperature and relative humidity. At temperatures above the boiling point (96°C), dry oxidation will occur, and there will be a thin, discontinuous water layer. At temperatures less than 96°C, there may

be a high vaporization rate (depending on the temperature and relative humidities). If there is a high vaporization rate, dry oxidation will continue to occur, as will thin, discontinuous water films. As vaporization rate decreases (thermal output of the fuel decays), water films will form that may be either thin (~0.1 μm) absorbed films or thick (>0.1 μm) films, depending on factors such as infiltration rate, relative humidity, and temperature. The rate of dissolution of the fuel matrix probably will not be limited by the thickness of these films.

The fraction of the fuel that is wet and available for aqueous dissolution processes ranges from a high limit of all the grain boundaries (assume a 10 to 30 μm grain size) to a nominal estimate provided by Gray and Wilson (1995) of a factor of 18 times the geometric surface area, or 3.9×10^{-3} m^2/gm. This wide an uncertainty arises because some mechanisms, such as mechanical disintegration due to mechanical loads and oxidation of the grain boundaries, may lead to increases in surface area with time. On the other hand, development of alteration layers may reduce the amount of reactive surface area. But if credit is to be taken for this process, the persistence and adherence of the alteration layers must be studied. Evidence to date from drip tests at ANL would indicate such layers are not reliably adherent to the fuel surface (Wronkiewicz et al, 1992, 1996).

I am not sure that the surface area, or the intrinsic dissolution model itself, is important to repository performance expressed as peak dose rate over a wide range of net dissolution times (i.e., time for complete oxidative dissolution of fuel). TSPA-95 showed non-sorbing, long-lived nuclides such as I-129, C1-36, and Se-79 as significant contributors to dose. Of far greater importance to the release rate of such nuclides is the instant release fraction from the gap. To illustrate this, assume that it takes 10,000 yr to dissolve the fuel; therefore:

$$(1/10^{-4}) \bullet (\text{percent in matrix}) = \text{percent released/yr}$$

assuming that 98 percent of the volatile fission products are in the matrix, 9.8×10^{-5} /yr will be released from dissolution. On the other hand, 2 percent or 2×10^{-2}/yr will be

released initially from the gap. Far-field dispersion processes, if they can be relied upon, will broaden, lower, and mitigate the instant release fraction pulse, as well as the pulse release in these same nuclides if a 1000-year dissolution time is assumed. The consequence is that peak dose rate is not particularly sensitive to net dissolution time of the fuel matrix, unless it is shorter than 1000 years.

The characteristics of failures of cladding and the container may limit the transport rate of reactants into and nuclides out of the pin/container. However, information is needed on the number, location, dimensions, and porosity/permeability of "failures"; time-dependencies; and the possible simultaneous failure of both container and cladding. Mass-transfer models of such "pin-holes" have shown significant reductions in calculated release rates of all nuclides. It is extremely doubtful whether reliable data related to these factors can be obtained over repository time scales.

There is also some uncertainty regarding TSPA-95 models for diffusive and advective transport out of waste packages into assumed water films flowing across the container surface. In particular, it will be important in TSPA-VA to credibly defend assumed boundary conditions such as film thickness, flow rate, and concentration gradients.

INSTANT RELEASE FRACTION

Having reviewed the recent AECL review for SKB (Johnson and Tait, 1997) on instant release fraction (IFR), I suggest that the Yucca Mountain project revise the 2 percent estimates for IFRs contained in TSPA-95. Of particular consequence might be the value for Cl-36 (with a revised value as high as 10 percent) and Tc-99 (with a revised value of 0.25 percent). Other nuclides of consequence, such as Cs-135, I-129, and Se-79, should be set at a revised value of 3 percent.

INTRINSIC DISSOLUTION

The intrinsic dissolution model expresses the forward reaction rate of the fuel matrix. Although we would like a mechanistic understanding of the processes, it is not necessary. We know that temperature, redox, and CO_3 are likely to be most important; Si , Ca, and burnup are likely to be secondary. At present, the dissolution model is less uncertain than are the inputs. There may be some advantage to the transparency of the modeling approach in paring back the number of independent parameters and examining the remaining parameters through sensitivity. Also, some of the data should be taken out, and "blind" sensitivity tests should be conducted to examine how well the model predicts the new, or "blind," data.

I suggest that the regression be conducted using only temperature and burnup as independent parameters. The model should be based on a threshold value of Si and Ca in Yucca Mountain pore waters (i.e., dilute). The correlation between the two should be included, as well. Assuming that burnup will be known within any given waste package, temperature and burnup will be true variables in the system, whereas pH, O_2 , etc., are properties of a time-varying environment. I suggest that the low O_2 data be excluded because they are not relevant to Yucca Mountain. Otherwise, all of the data can be included in the regression, and the "scatter" in the regression can be treated as variability. Then variations in individual components (e.g., spent fuel vs. UO_2, pH) can be tested in sensitivity analyses. Finally, blind tests can be conducted by excluding some of the data and using the regression to see how well those data are predicted.

STABILITY OF SECONDARY PHASES

The long-term mobilization rate of most nuclides (with the exception of I-129, Cs-135, and Sc-79) from the fuel (and glass) matrix is expected to be limited by radioelement solubilities rather than dissolution rate. In reviewing the solubilities "elicited" in TSPA-95, the uncertainties probably derive from uncertainties in determining which phase will form, its crystallinity, and the degree of co-precipitation of trace nuclides. Further, these

estimates, which are based on oversaturated tests, follow from high concentrations to lower. A more complementary approach is to start at subsaturation and slowly increase concentrations of all nuclides consequently to examine what solids precipitate, as well as which nuclides substitute (co-precipitate) into these alteration solids and to what extent. Current ANL drip tests show evidence of this effect for Np, but concentrations are at the ragged edge of detectability. Similar results were observed for Tc released from glass in the 1980s. Longer tests may not resolve this process. I suggest that drip tests be conducted that titrate the element of interest, then look at the concentrations that develop as a function of time. In this way intermediate phases will be identified and, after the concentration no longer rises with additional titration, a true solubility limit for the element can be established. Recent attempts to revise Np solubility limits using waste-form tests are a first step in this direction. This early attempt with Np shows Np concentrations from waste-form tests considerably lower than previous pure-phase solubility tests. The concern is, however, that the concentrations from waste-form tests may not reflect active incorporation of Np into alteration products (hence a fixed co-precipitation concentration), but rather are still slowly increasing with time.

For the TSPA-VA, I do not think it is reasonable to combine the "elemental" solubilities from TSPA-95 with those "effective" solubilities coming out of the recent low drip-rate and vapor tests (discussed in presentations by A. Wijesinghe and W. Halsey at WFEE Workshop 2), which in most cases are far lower. They should be kept as separate conceptual models: elemental solubilities versus "effective" concentrations. It is important to develop a good understanding of the uncertainties in the new test results. Right now the rest of the world uses elemental solubilities. To rely on effective solubilities, considerations of thermodynamics and predicted phases must be merged with the phases that actually are seen. Plus the solubilities of the secondary phases must be assessed; the use of natural analogs may help support the rates. I suggest that the rates derived from the new data be used in sensitivity analysis, but not be relied on at present.

To estimate the dissolution rates that should be used when all of the UO_2 has dissolved, one needs information on the alteration products that will exist and the element content

that they are expected to have. This information will provide an initial concentration of various nuclides that are available for mobilization through dissolution of the secondary phases.

The above discussion and recommendations on spent fuel regarding testing of co-precipitation solubility limits also pertain to the dissolution/alteration of the borosilicate glass waste form. Evidence suggests that elements such as Tc and Cs, rather than displaying extremely high solubility limits, are incorporated into glass alteration products such as clay and disordered zeolite.

COLLOIDS

The issues associated with radionuclide-bearing colloids are: Do they form? Are nuclides irreversibly bound? Are they stable? Are they transportable? The transport issue is probably the most important from the viewpoint of eliminating this potential near-field release mode. My bias says that the colloid issue is more important in the near field than in the far field because the concentration of nuclides is highest in this region.

Three types of colloids must be considered: degradation colloids coming from the alteration layers, true precipitation colloids coming in with the groundwater, and other colloids related to surface sorption. It is difficult to imagine degradation colloids going far in film transport (without advective flow). Degradation colloids in advective flow conditions are difficult to model because there are no data on agglomeration, chemical dissolution, or filtration. Colloids reflect a non-equilibrium condition. At most, all of the alteration products are potential colloids, especially in the presence of advection due to dripping, and their transport should be evaluated.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

I suggest that there is a misplaced emphasis on reducing uncertainty, rather than seeking to increase confidence. The two are not necessarily the same thing. Adding speculative

processes and conditioning models to data from analog systems of dubious relevance is not a proscription for attaining either professional or public approbation. Instead, I suspect that transparency, peer review, and public acceptance will be enhanced if reasonable bounding models and data are applied. For example, it is conceivable that irradiated cladding may persist in an unfailed condition for many thousands of years within a repository; but it is more than conceivable that we cannot credibly sustain such a prediction within a rigorous licensing process in the face of variable future repository conditions, uncertain scenarios, and models largely predicated on short-term corrosion tests of non-irradiated cladding.

Perhaps even more to the point for the TSPA-VA is: why add such a problematic process when it apparently has no effect on the predicted peak dose rates of long-lived radionuclides from a repository at Yucca Mountain?

An immediate recommended action to reduce uncertainty is to stop confusing uncertainty (a property of the analyst) with variability (a property of the repository system). Ignorance and uncertainty cannot be disguised as expected variability that may spread and dilute the source-term release of radionuclides from the near field.

Another recommended action with respect to source-term modeling is to stop relying on future properties of failed engineered barrier materials (container, cladding, waste form) that are assumed to be and remain favorable during the repository lifetime. Ad hoc, favorable assumptions regarding the timing, location, and characteristics of water contact mode with the waste form also should be avoided in any defensible safety analysis. Unfortunately, by excluding unwarranted optimistic assumptions, I see no way to effectively reduce uncertainties in transport of aqueous and colloidal fractions of radionuclides in the current engineered barrier system (EBS) design. Yet it is these transport uncertainties, more than any other factor in the near field, that have led to unfavorably high release rates of radionuclides from previous performance assessments.

The issue of colloids generated at the waste form surface and migrating into the geosphere was not considered in TSPA-95. Without design changes, I see no way to exclude this possibility of radionuclide mobilization, which could have a significant, adverse impact on release rates from the near field. Elimination of the potential colloid release mode might be based on showing that (a) colloids cannot form (which is different than showing that colloids do not form under a specific set of experimental conditions); (b) colloids are not stable; (c) colloids do not irreversibly sorb radionuclides. I doubt, however, that any of these issues can be favorably resolved as easily and as reliably as introducing a barrier to prevent colloid transport in the near field.

Further development and extension of the oxidative dissolution model for spent fuel ought to be postponed until a rigorous sensitivity analysis of this parameter is made with respect to meeting a dose-rate criterion. Analyses (my presentation at WFEE Workshop 3; Pigford, in press) seem to indicate that peak release rates from the near field for dose-contributing nuclides are dominated by the instant release fraction, and that dose rates measured at some distant, accessible boundary are insensitive to large variations (several orders of magnitude) in the dissolution rate of the spent fuel matrix.

A closer connection should be made between solubility studies/thermodynamic estimates of radioelement solubilities and the measured radionuclide concentrations from waste-form alteration tests. The recent Neptunium work by D. Sassani (presentation at WFEE Workshop 2) and others for the Yucca Mountain project is a good step in this direction. However, the uncertainty band between measured solubility concentrations and measured concentrations from waste-form alteration tests is still too wide (many orders of magnitude).

Introducing water that already contains trace (but measurable!) amounts of dissolved radionuclides into existing drip-style tests might promote the growth of alteration products that contain measurable amounts of trace radionuclides. Both the identification of trace radionuclides in alteration phases, and the measurement of reduced concentrations of radionuclides in the exiting water, would help to narrow the uncertainty

bounds between solubility measurements and waste-form tests. It would also confirm active removal of trace radionuclides from solution into alteration phases. Confidence in this activity would also benefit if an a priori co-precipitation model is used to make blind predictions of reduction factors in concentration for several radioelements between ingressing and egressing water, and these predictions are then compared against the tests results.

Finally, the issue of Iodine volatility should be resolved by direct spectroscopic measurement of Iodine isotopes obtained by vacuum-pumping the gas volume from freshly breached, cladded fuel rods. Perhaps such evidence already exists and can be utilized, rather than initiating a new R&D program.

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ELICITATION SUMMARY
JOHN K. BATES

OVERVIEW OF WASTE FORM DEGRADATION PROCESSES

The waste form degradation process is one of many that must be evaluated to assess the total system performance of a repository. In the past, there have been disconnects between the various parts of the total performance assessment that are just now being addressed with the TSPA-VA. Some of the disconnects stem from the lack of a common connecting thread that could combine input from the major parts of the project, e.g. repository, hydrology, and materials, in a manner that makes common sense. One standard for predicting the long-term behavior of materials in a repository environment is ASTM C-1174. This standard can be applied to materials, but also all aspects of repository performance, in that it is a common-sense guide to combining modeling and testing to make predictions. As applied to waste form degradation, the standard requires that the first information for performing long-term predictions be an understanding of the environment and identification of materials. Although these issues are not completely defined, enough is known to have initiated reasonable waste form tests more than 10 years ago. These tests address the "most likely" conditions that will be present in the repository, bounding conditions, and conditions that may accelerate the reaction of materials. Realistic assessments require that in-service conditions be evaluated to provide that all reaction processes are identified and addressed. These tests can also become tests to validate modeling if they are run for long periods. Response tests are performed to understand the mechanism that controls the degradation rate as different physical parameters are varied, and accelerated tests help validate that the mechanisms observed in in-service conditions and response tests are rate-controlling as the reaction reaches advanced stages. In applying this approach to the unsaturated environment at Yucca Mountain, one recognizes that the repository will be basically dry and that immersion, although possible, is not the expected condition. This situation differs from many other

repositories being studied worldwide, so that many of the preconceived ideas about materials behavior may not apply to the Yucca Mountain environment. Thin films of water in a vapor environment represent the expected condition—or worse than expected—for the degradation of materials, rather than flowing and abundant water. However, thin films of water may limit the transport of radionuclides, compared to a repository that has a greater amount of water.

CLADDING

As recommended in the Yucca Mountain Site Characterization Plan many years ago, evaluation of spent fuel cladding should begin with an examination of failure modes. General corrosion will occur, but at very low rates. Localized corrosion is likely to be the most damaging process that affects the cladding. Contact points between the rods and the support structure could serve as crevices for the initiation of localized corrosion. If waste packages are placed close together (as in a line-load configuration), high Cl- solutions developed within waste packages containing high-level waste (HLW) glass could mix with spent-fuel packages and enhance localized corrosion processes. Hydrogen embrittlement and hydride reorientation of the cladding can weaken it to the point that mechanical loads, such as from earthquakes or rockfalls, could lead to failure. The operation of all of these failure modes will lead to a distribution of failures within a package and across all packages.

No tests have been performed on failure of cladding. For example, corrosion tests of irradiated Zircalloy have not been performed. In the absence of these tests, one must rely on non-qualified data to take credit for cladding. The most likely conditions (e.g., water chemistries, earthquake loading) should be identified and tests conducted. Given the present lack of data, I believe that TSPA should not take credit for cladding when modeling the time from failure of the waste package to dissolution of the waste form.

SURFACE AREA EXPOSED AND WATER CONTACT

It is important to define what is meant by "exposed" surface area. From the first failure of the cladding, all of the spent-fuel gap region will be exposed to the environment (water vapor, air, oxygen, carbon dioxide, and so on). This environment will depend on the time of failure. For all environments, the rapid release fraction from the cladding gap will be mobilized immediately. The fraction from the grain boundaries will be mobilized soon after contact with water vapor. This in effect happens because, in a humid-air environment, all surfaces in contact with the atmosphere will immediately develop a thin water film. Reaction along the grain boundaries will follow shortly thereafter, and any soluble material in the grain boundaries will be released. Tests with spent-fuel fragments and water vapor demonstrate that the grain boundary regions react. However, to date the vapor tests do not show the same reaction that occurs on the outer surfaces of fragments. On the outer surfaces, dissolution and alteration result in the formation of hydrated uranaly phases. On the inner surfaces, dissolution/oxidation occur, but there is no evidence for hydrated phases. When spent fuel reacts with groundwater, the same type of reaction progress is observed, but phases containing cations (e.g., Na boltwoodite) are observed on the surfaces, but not in the grain boundary regions. It is not yet known whether the same processes will occur with clad fuel, and if they do, whether the formation of secondary phases will result in splitting of the cladding. The flow-through tests are designed to measure a forward reaction rate of the fuel; these tests give the rate at which the fuel surface reacts under the test conditions. If the test is run such that the surface reaction is congruent and there is no precipitation of material on the fuel surface, the highly soluble species such as Tc will be released at the rate the fuel reacts. Flow-through tests can give different forward reaction rates as a function of test conditions. Care must be taken to understand the use of the data in modeling the reaction of fuel under high-flow, dripping water, and water vapor conditions, to clarify the mechanism by which the fuel is reacting is clear.

A realistic model of surface area would account for the exposure conditions and short-term and long-term processes, considering also the 100-year time steps of the TSPA. If

the fuel is exposed immediately to small volumes of water (as at a site exposed to dripping conditions), then the area available for release in the first year would be the gap inventory plus the geometric area plus the fragment surfaces. Within the first 100-year time period, most of the grain boundary surface area also would be exposed. More likely conditions during the first 100-year time step are that the fuel first would be exposed to water vapor; reactions to the vapor would then form an alteration layer; then flowing water might follow. In this case, a short-term surface area would be the geometric area plus the fragment surfaces plus an alteration layer on grain boundaries (~ 1 um). A simplified long-term surface area should include the geometric area of all the grains. There should be no credit for limiting the exposure to vapor-phase moisture (in the presence of high humidity and absence of drips), because the same aqueous processes should be operative over the long term.

It is reasonable to include in the models both advective and diffusive pathways, and to take credit for the fraction of the surface area that experiences advective and diffusive flow. Alteration phases between the fuel and the cladding may make the diffusive pathway more convoluted or may have no effect. Which result will occur is unknown at this time. There will be a constant concentration of highly soluble species such as Tc along the length of a rod in the saturated, porous alteration layer. An advective pathway that intersects part of that rod will remove the Tc at this concentration. Diffusive transport will then move Tc into the advective pathway to maintain the concentration. This process will provide a concentration limit for highly soluble radionuclides in low-flow conditions.

INTRINSIC DISSOLUTION

It is important first to define what is meant by intrinsic dissolution and to distinguish it from forward reaction. Intrinsic dissolution can be defined as the process that occurs in the presence of de-ionized water, which can be measured as a function of temperature, proving that the reaction mechanism remains constant. With this definition, the rate of

intrinsic dissolution is a property of the material itself. A forward reaction rate, on the other hand, depends on a variety of conditions such as groundwater composition, flow rate, and back-reactions. The present dissolution model accounts for variations in the intrinsic rate by using forward reaction rates to account for physical parameters that may vary during the repository disposal period. The rate measured in tests in which the intrinsic reaction rate is measured should give the same reaction rate. Thus, if the reaction rate in a drip test depends only on the reaction between spent fuel and water, or radiolysis products of the water, then the rate should be similar to the rate measured in a flow test performed at the same temperature with the same water conditions and radiolysis products.

The present intrinsic dissolution model for spent fuel is a defensible approach. It is also useful to incorporate a range of data on pH, burnup, HCO_3, Ca, Si, etc., to account for the effects of these parameters on dissolution rate. It probably is wise to separate out the dissolution data for UO_2 and spent fuel in the database for the dissolution model. In drip tests with UO_2 and spent fuel, although the same sequence of alteration products is observed, there is a difference in the reaction that occurs at the fuel surface and in the grain boundaries. This difference may be due to radiolysis or composition. The temperature dependency that is used appears to be reasonable.

The glass dissolution/reaction rate has been determined under static and dripping conditions, but not in vapor, as a function of temperature. Under static conditions, the glasses tested all undergo a reaction progress in which the glass initially reacts rapidly (forward rate or Stage I); the rate slows as the amount of Si in solution increases (Stage II); and finally, when alteration phases form, the rate increases (Stage III). The rate under dripping water conditions appears to be near the Stage II rate, but the release of transuranic elements is congruent with the glass dissolution, and Pu and Am are released as secondary phases incorporated into clay colloids. As with spent fuel, the glass initially will contact water vapor. The glass is very hydgroscopic and will become covered with a film of water. Thin film reaction will occur, whereby the reaction rate quickly reaches Stage III. At present we do not know the thin film hydration rate as a function of

temperature. This information would allow a calculation of the amount of hydration that would occur before contact with dripping water. For TSPA-VA, we can provide a look-up table for the hydration rate based on a limited number of tests with early glass compositions. It is likely that the glass will hydrate quickly; after several thousand years there should be a fully hydrated volume. The importance of reacting hydrated glass with dripping water is that all elements not incorporated into secondary phases during the hydration process are immediately released from the volume of hydrated glass. Tests indicate that these include Pu, Am, high concentrations of anions (Cl^-, etc.), and soluble glass components such as Na, B, and Li. Thus, Pu and Am can be released from the glass as a pulsed soluble component, or more slowly, but congruently with the glass reaction, as colloids. Activity related to the modeling of glass reaction peaked in the late 1980s, and little advance has been made to incorporate such features as described above into submodels that provide information to TSPA-VA. Also, considerable short-term information on glass consistency is available from the waste producers. A connection must be made between this short-term data and long-term behavior.

RAPID RELEASE FRACTION

The rapid release fraction from the reaction of spent fuel will consist of the gap inventory and in part the grain boundary inventory. For commercial fuels studied in the Yucca Mountain project, this release comprises a small fraction of the total inventory of elements such as Np and Tc. Additionally, if the fuel reacts at the rates reported for drip, flow-through, or semi-static tests, a rapid release fraction loses much of its meaning because the reported rates are rapid enough to react most of the fuel within a few thousand years, depending on temperature. It should be noted that although the fuel reacts, not all of the radionuclide inventory may be mobilized. In fact, there is convincing evidence that many elements are incorporated into secondary phases. This process likely will retard radionuclide release to some degree. The overall credit that can be provided by secondary phases has not been evaluated.

For glass, the rapid release fraction will depend on the extent of vapor hydration that occurs prior to contact with dripping water. Although the inventory of Pu and Am in the glass is much less than that in the fuel, the concentration of Np in glass is similar to that in fuel. TSPA calculations must not overlook the ramification of a rapid release from glass, both on the release of actinide elements and on glass components that will affect the near-field environment.

STABILITY OF SECONDARY PHASES

Tests show that the fuel reacts quickly and forms alteration phases. Natural analogs, calculations, and tests with UO_2 all indicate that the same sequence of secondary phases form. Thus there is high confidence that these phases will form under unsaturated disposal conditions. There is some uncertainty as to the final assemblage of phases that will form, due to potential changes in the groundwater composition and the kinetics of phase transformation. But the expected conditions will result in the formation of the same major mineral groups. We do not expect perfect end-members like those found in solubility studies.

A key question is what the ultimate effect of the phases will be on the release of radionuclides from the waste package. Alteration and volume expansion may lead to splitting of the cladding, immediately obviating any protective effect that may be given to cladding. The alteration phases may also incorporate actinides such that the phases must redissolve to be mobilized, or they may sorb radionuclides to their surfaces. Alteration phases may form within the fuel but not split the cladding, thereby forming a barrier to advective flow. Thus, even though the fuel may continue to react, the release of radionuclies could be reduced. Because of the large surface area of the phases, they may form a tortuous pathway for radionuclide transport and thus delay diffusive transport. It is unlikely, however, that the formation of alteration phases will result in a reduction of the reaction rate of the fuel. Tests to date, in which an alteration layer as much as 40 μm

thick has formed on the surface of the fuel, do not demonstrate a conclusive retardation of the intrinsic rate.

It is important to evaluate the effects of groundwater on alteration phases and to look at the degree to which various radionuclides are incorporated in the secondary phases. In the high-drip-rate-tests, it appears that the fuel passes through intermediate phases (e.g., schoepite) so quickly to stable phases (Na-boltwoodite) that the intermediate phases have not been observed. Less Np emerges than Tc, although both are soluble. The Np must be held up by some process(es). Although there are no calculations that indicate that Np can be incorporated into the boltwoodite structure or should be sorbed on the surface, it is reasonable to expect that either could happen. It is possible, using calculational techniques, to predict whether uranium phases can incorporate other actinide elements into their structure. Such calculation could add credibility to observations from in-service condition tests. After identifying secondary phases that can retain radionuclides, additional tests can be performed to evaluate the stability of those phases. In tests currently performed with dripping water, a substantial decrease in the release of Pu is observed with high confidence that it is being held up; we also see a decrease in Np, but are not sure how it is being held up. Other important radionucles are also incorporated into secondary phases.

In terms of a retention factor or effective solubilities for various radionuclides, we can consider the credibility of the drip tests (limited in number) and the old data that were used to estimate solubilities for TSPA-95. The old data were based on supersaturated conditions, and often collected in the presence of carbonate. The drip test results are more credible because they are based on expected conditions, but it is not yet possible to rely on them completely because we have not demonstrated where the elements of interest reside within the alteration phases. Pu and Np should have the highest priority and the highest potential for providing estimates of effective solubilities. In the absence of dripping water, Np goes into dehydrated schoepite, which is known, under certain oxidizing conditions, to be stable for long periods. We see preliminary evidence of both Pu and Np in alteration phases. We need to examine more closely the ways their initial

secondary phases alter and the disposition of actinides during the subsequent alteration process(es).

At present, I recommend that the effective solubilities for all radionudlides be assessed by weighting the drip test results heavily, but incorporating the old range of solubilities from TSPA-95 as a low-probability tail of the distribution. To completely revise the effective solubilities, calculations and experiments must be conducted that demonstrate how the radionuclides are incorporated into or adsorbed onto alteration phases.

Sassani and Siegmann (1998) presented a rationale for using a lower range of Np solubilities in TSPA-VA based on information from a range of experiments (references distributed to expert panel). This paper is a good place for the staff responsible for the TSPA-VA to start reviewing the data from different tests and using these data in place of previous data, that, while applicable to select situations, are not broadly applicable to repository disposal conditions. With additional work, it may be possible that the range for Np and other elements can be reduced to lower values.

COLLOIDS

The colloidal state has several properties. Colloids are particles having sizes that range from 1 nm to 5 um. They are surface-active agents and dispersed particles. Both their physical and chemical characteristics govern their stability, adsorption properties, and behavior. Their properties depend on the environment and the composition of the waste form. We are interested in characterizing colloids by their radionuclide-bearing phases, their growth and agglomeration, their diffusion properties, their stability, and their sorption properties. I place radionuclide-bearing colloids into three categories: pseudocolloids, which are found in groundwater and sorb radionuclides; real colloids, which are polymers of soluble radionuclides; and primary colloids, which are derived from waste form degradation.

With spent fuel, the amount of actinides initially released as a colloidal fraction is about 50 percent of the total release. This percentage may reflect fines being rinsed from the fuel surface or reacted fuel grains being carried due to advective flow. As secondary phases form, the percent of actinides in colloids decreases as the secondary phases block direct advective flow pathways from the fuel and cover the fuel with a mat of material. As more secondary phases form, their volume increases, and they may not be supported as part of the original structure but may spall off the reacted fuel. There is evidence that spalling is occurring in some of the drip tests. When this occurs, the fraction of actinides released in a colloidal fraction increases.

Tests have been conducted on both non-hydrated and hydrated glass. Drip tests on non-hydrated glass show that initially a gel layer forms on the surface, and then becomes a clay. With continued dissolution of the glass, the clay layer breaks off into small pieces that have a plate-like shape. In water vapor, the secondary phases include calcium silicates and zeolites, as well as clays. When water is dripped onto hydrated glass, a brine solution is developed that has a high concentration of anions. All of the ions not incorporated into secondary phases are washed out along with the actinides. I am concerned that this chloride-rich solution may leave a HLW-glass container and, through capillary flow, make its way to a spent-fuel container. The solution could cause rapid localized corrosion of the Zircalloy cladding. Sensitivity analyses should be conducted to examine this possibility as a function of the distance between containers. We plan to study the hydration rates of glass, but I would expect most of the glass will be hydrated rather quickly. In sum, the tests to date show that Pu and Am are readily mobilized and transportable from glass, both as a 100 percent colloidal fraction or as 100 percent soluble, depending on the condition of the glass. The potential for transport out of the container and the potential for their filtration should be evaluated.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

- Identify the conditions of temperature, water availability, air composition, and groundwater composition that will exist in the waste package environment as a function of time.

- Identify materials that will be in the waste package and finalize the waste package design.

- Identify the major modes (see ASTM 1174-C) for corrosion of spent fuel and glass.

- Identify the key physical parameters that affect corrosion of glass and spent fuel, decide for which parameters enough information is known for input to a license application, and expedite work to get information about those that require more information.

- Agree on submodels for glass and spent-fuel corrosion; convince anyone, someone, that they make sense, and demonstrate that they are consistent with the data and have predictive capability, then use them in the TSPA effort.

- Determine how to account for radionuclide release as affected by secondary phase formation for both glass and spent fuel.

- Address key issues related to cladding credit.

- Address key issues related to spent-fuel corrosion: the effect of radiolysis on corrosion rate, the stability of phases as a function of changing disposal conditions, the effect of secondary phases on cladding and spent-fuel integrity, the mechanism of spent-fuel reaction, and the effect of burn-up on radionuclide release.

- Address issues related to integration of components of the engineered barrier system (EBS) with waste form reaction, with respect to both incoming water composition and transport of radionuclides and colloids through the EBS.

- Address the information needed about primary colloids: what they are, where the radionuclides are located, their physical properties, and whether they can be transported.

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ELICITATION SUMMARY
WILLIAM BOURCIER

OVERVIEW OF WASTE FORM DEGRADATION PROCESSES

Water contact scenarios will be very important to waste form degradation processes. Whether the water is vapor or drips and, if drips, how much water flows into and through the waste package will be significant to various processes. The water contact scenarios should be coupled with the thermal history of the waste packages. For example, a hot, oxygenated environment could be a strong driving force. Carbonate brines, which may develop through evaporation, could dominate the dissolution processes. It is also important to know the persistence of various environmental conditions. A related issue is the potential, soon after breach of the waste package, for oxygen to be driven off, which could be important for the fuel oxidation process (Shoesmith et al., 1997). The modeling needs to better represent the coupling between hydrologic conditions and chemistry.

Another significant process is the interplay between advective and diffusive flow. We need to know what fraction of the waste packages and of the waste form will be involved in aqueous processes. Should we assume that all surfaces are wetted? That all surfaces are exposed to a drip flow path? Alteration products tend to fall off and can be transported in an advective flow path along the bottom of the canister. We should attempt to calculate the thickness, porosity, and tortuosity of alteration products and use the result to better estimate diffusive flow rates.

CLADDING

Degradation processes for spent fuel cladding are not in my area of expertise. There appears to be general agreement on processes related to hydrogen buildup from hydrolysis leading to embrittlement in the absence of water. Likewise, hydride formation and the potential for hydride reorientation have been studied. But we do not have good

models for the modes and rates of cladding degradation after water becomes available. General and localized corrosion are possible, and microbiologically influenced corrosion is potentially important. Also, models for the influence of mechanical loads on the cladding do not appear to be well developed. Because of these uncertainties, I think it is not at present prudent to take credit for the integrity of cladding.

The pour canister for high-level waste (HLW) glass is 304L stainless steel. It is possible that the microstructure of the stainless steel will become sensitized over long periods, such that it is not appropriate to take credit for the canister in considering degradation of the glass.

SURFACE AREA EXPOSED AND WATER CONTACT

Glass

Studies by Wicks (1985) examined the surface area of fragments formed from cracks in glass. He found a rate of release that implies a surface area of 20 to 25 times the geometric area of the glass log. Retardation mechanisms include the development of clays and zeolites, which develop rapidly at 90°C. These would, if anything, limit the access of water because of swelling and thereby decrease the effective surface area as more clay develops.

In a humid atmosphere, a water layer will develop on the glass that is tens of Angstroms to a few nanometers thick. As an alteration layer develops, the thickness of this water layer will remain constant. With this water layer, the reaction rates will be approximately the same as under saturated conditions. Under humid-air conditions, vapor hydration will occur. It is possible that vapor hydration could occur for some time and be followed by episodic advection (perhaps caused by climate change). Such conditions could lead to a pulsed release. The process could be modeled using a long-term aqueous glass reaction rate and assuming all of the thickness developed can be washed away by advective flow.

Spent Fuel

For spent fuel, the issue is how much oxidation has taken place and how many grain layers are available for dissolution. The answer depends on the temperature and oxidation history. Flow-through tests suggest a factor of 15 times the geometric area of the fuel (presentation by W. Gray at WFEE Workshop 1). I conclude that a reasonable surface area for the dissolution model is the geometric area plus the fragment surfaces plus 5 to 10 grain diameters. Similar to glass, the development of secondary phases probably will lead to a long-term decrease in this surface area for spent fuel.

INTRINSIC DISSOLUTION

The intrinsic dissolution or forward reaction rate of glass depends primarily on pH, SiO_2, and temperature. The incoming water is essentially benign; expected pH is near neutral, and the groundwater is already saturated in SiO_2 from having passed through the silicic tuffs, which contain silicate glass. There undoubtedly will be some variability in SiO_2 conditions, but the relevant long-term dissolution rate is for SiO_2 saturated conditions. Two measured values for the dissolution rate of SRL-202 glass under silica-saturated conditions are 0.0016 and 0.0025 $g/m^2/day$. Note that these rates do not include retention of alteration products. To do so would require knowing the dissolution rates of secondary phases such as smectites and zeolites, which will have to be addressed after the TSPA-VA. To estimate the vapor hydration rate, it probably is conservative to use the Stage III reaction rate. This higher rate would produce a higher alteration rate and a thicker alteration layer.

The NRC noted that the forward reaction rate for glass may be non-conservative if radionuclides migrate to the surface (presentation by T. Ahn at WFEE Workshop 2). However, the diffusion rates in glass are very low, and this migration process will be slow. Further, if diffusion were to occur quickly, the migration products could crystallize at the surface and slow the mobilization.

The models of forward reaction rate proposed for spent fuel are all semi-empirical models based on a macroscopic thermodynamic approach. None are mechanistically based on a molecular level. No one has approached the process from first principles, for example based on the molecular structure and electrical properties of the oxide/water interface. For this reason it is not fair to reject or favor a theory based on one model being mechanistically based. All the models end up being regression fits to experimental data, with the form of the equation consistent with a macroscopic, thermodynamically based model. Given the uncertainties in the data, the differences in the regression models will be insignificant. I believe the present intrinsic dissolution model is reasonable. It does not, however, include the Ca - CO_3 effects examined by Shoesmith et al. (1996). Retaining a burnup term may be useful for later sensitivity analyses that will allow for radiolysis and fission products to be considered in an empirical way, without having to consider the mechanisms. Alpha radiation is likely the most important radiation effect on fuel dissolution and does not vary much with time.

RAPID RELEASE FRACTION

The rapid release from spent fuel is outside my area of expertise. However, it may be important to consider an analogous process for glass. It may be possible to develop a reservoir of radionuclides from vapor hydration that are then washed away suddenly by a pulse of water. The amount washed away at any given time could be assumed to be all of the alteration products developed at the forward reaction rate.

STABILITY OF SECONDARY PHASES

Developing secondary phases and establishing effective solubilities is important to realistic estimates of radionuclide mobilization. The techniques for evaluating speciation and solubility are available, but there are limited data on which to base an assessment.

Glass

For glass, an average retardation factor can be used to represent ion exchange that retards the transport of radionuclides. There is evidence that Cs and Sr exchange into clay phases (OCRWM, 1997), but the work has not been abstracted, nor is it being pursued. A model is needed for the dissolution of the clay alteration products after the glass has been entirely altered. But it is important to remember that conditions must change for the system to go out of equilibrium. If the system remains in equilibrium, the clays will not dissolve. Therefore, environmental conditions must be tied to the near-field stability model, and the stable phases must be identified. One can use sorption tests to look at what is removed, then assess partition coefficients, expressing the secondary phases in the glass and the concentrations of the solutions. Sorption coefficients, which have been assessed for crushed tuff samples, can be used to evaluate the retardation that occurs along flow paths.

Spent Fuel

For spent fuel, an important issue is the degree to which the new drip-test data (Finn et al., 1996, 1997) can be used to estimate effective solubilities (including secondary phases), replacing the solubility estimates based on the expert elicitation in TSPA-95. The spent-fuel drip tests, although experimentally simple, are actually complex experiments. Multiple simultaneous coupled physical and chemical processes take place, which are only partly constrained by the data obtained from the test (the solution analyses). Therefore, numerous assumptions must be made to arrive at a process or mechanistic information. I think it is mandatory that the relevance of these service-condition tests for predicting repository behavior be validated. We need to determine whether the processes that control radionuclide concentrations in the thin water film of these small-scale, short-term tests are the same processes that will control radionuclide concentrations in the repository. Given this uncertainty, I think it is premature to use the results directly in PA without strict qualification. For example, I would like to see some mechanism proposed to explain why Tc is released much more slowly than it would be if it were released stoichiometrically (as fast as the UO_2 is being altered). Will this same mechanism delay the release of Tc in the repository? I realize that experimental work to

acquire these types of data is in progress at Argonne National Laboratory (ANL) (personal communication, John Bates), but it is not yet available. Therefore, at this time I cannot recommend using the results for estimating spent fuel release rates in TSPA-VA.

The approved and reasonable methodology to make long-term performance predictions calls for developing a mechanistic basis with appropriate experiments and modeling, then testing (validating) the model using service tests such as the drip test. I cannot see being able ever to justify PA models based entirely on analyses of complex service-condition tests. Although A. Wijesinghe's work (WFEE Workshop 1) on interpreting these tests is first rate, there are too many unknowns and too few equations to make the interpretation anything more than one among many. We should be able to predict the results of the drip tests using the results of the flow-through test on spent fuel, with an appropriate saturation (affinity) term coupled with predicted secondary phase formation that includes the co-precipitation of radionuclides such as Np along with U in the phases. With this sort of mechanistic basis for understanding the tests, we can then fill in the rest of the unknowns (primarily involving water contact modes) with reasonable guesses and see whether we come close to predicting the solution concentrations in the residual fluids. This sort of validation should be the primary use for the drip tests.

With regard to the revision of Np solubility data (presented by D. Sassani at WFEE Workshop 2), it should be noted that Yucca Mountain project technical staff have been recommending for 20 years that we measure the basic thermodynamic data for all the relevant radionuclides as part of any serious effort to make a geologic repository. It is straightforward work, and not expensive relative to project funding. We should not be in the position now of needing experts to guess solubilities (as was done for TSPA-95).

I believe D. Sassani makes a credible case for changing the distribution of likely Np solubilities to his recommended values (TRW, 1998). The high solubility values in the previous expert-recommended distribution are based on experiments in which highly supersaturated solutions were allowed to equilibrate. Because of the slow kinetics, probably having to do with reduction of NpO_2^+ to Np^{4+}, the solution did not precipitate

the thermodynamically most stable phase (NpO_2), but instead remained at metastable equilibrium with more soluble Np phases. Similar highly supersaturated conditions may not be possible in the potential repository, although there are experimental data for the analogous actinide Pu that indicate they may be possible.[1] The bulk of the data reviewed and reassessed strongly suggest that Np will co-precipitate with U in various uranium alteration phases. In addition, recent experimental studies of Np solubility in J-13 water at 200°C have shown that the precipitation of NpO_2 from oxidizing solutions is relatively fast and that we may be able to assume solubility-limited Np concentrations for at least some anticipated repository conditions (Wolery et al., 1998).

Np and U appear to be released congruently (that is, in the same proportions) from spent fuel in most of the tests. This result indicates the spent fuel simply dissolves, then quickly re-precipitates as uranium alteration phases (oxides and silicates) that incorporate the Np as well as U. Recent experiments in which Np has been found in the alteration phases of a reacted spent-fuel sample support this interpretation (see Buck et al., 1997). This is the most likely process that will control Np release from spent fuel in the repository. It is also consistent with the service-condition and single-pass flow-through (SPFT) tests. I believe the new recommended values are consistent with this mechanism, and with available experimental data.

One other potential problem comes to mind. In nature, rapidly precipitating mineral phases often contain higher concentrations of trace and minor elements than do phases that form more slowly, or the phases they will eventually age into. It is conceivable that the uranium alteration minerals could expel their guest Np contents at some later time when they recrystallize. I do not believe it is possible to infer with available data whether this will happen, or to give any rate estimates.

[1] In dissolution tests of Pu-containing ceramics, measured Pu concentrations in J-13-like waters were 10 to 100 times higher than solubility limits of $Pu(OH)_4$ and 5 to 10 log units higher than solubility limits for PuO_2. The Pu solution was primarily PuO_2^+. These tests were funded by the DOE Fissile Materials Disposition program.

Regarding the modeling of dissolution after the UO_2 is completely altered, remember that the system is now closer to equilibrium. The alteration minerals that form are now equilibrated with the solution phase. Therefore, the dissolution rate of secondary phases will be lower than before, and without a change in conditions will be essentially zero. It is possible that the Np will concentrate in the alteration phases. However, even if we assume that it all becomes $(U, Np)O_3$, the rate of dissolution will be less than the dissolution rate of UO_2. Therefore, I expect the mobilization rate to be lower than for spent fuel.

COLLOIDS

Clays developed from alteration of glass are a potential source of colloids. Although probably only a small amount comes off as colloids, 50 to 90 percent of Pu is mobilized as colloids. In cases where glass is mixed with spent fuel, the colloids from the glass could be a problem in that actinides released from the spent fuel will sorb onto the colloids generated by the glass alteration. In addition, ferric hydroxide colloids from iron oxide corrosion products will be present from oxidation of the waste package and internal support structures. These colloids will be available for transporting radionuclides from both glass and spent fuel. Changes in pH and the ionic strength of pore fluids in the near field will affect the mobility and stability of colloids.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

There are six areas in which key data are missing and for which activities that address these data needs are warranted.

- Thermodynamic solubility measurements are needed, especially related to Neptunium and the kinetics of NpO_2 precipitation. The question is whether NpO_2, with Np present as +4, will equilibrate under oxidizing conditions with Np+5 in solution. If we can assume Np is solubility-limited by equilibration with solid NpO_2, the long-

term Np problem for the repository essentially disappears. The same tests should be done for PuO_2, for which the same question arises.

The need for thermodynamic data also exists for U, Np, and Pu silicates. Some data exist for the uranium silicates. Essentially no data are available for Pu and Np. If PuO_2 does not equilibrate with oxidizing fluids, as appears to be the case based on experimental data, some other phase containing oxidized Pu will be the solubility-limiting Pu phase. Based on an analog with U, this could be a silicate phase. The same argument holds for Np. In addition, based on crystal chemistry, both Pu and Np are likely to be incorporated into uranyl silicate phases as trace or minor components. It would be useful to have data on the partitioning of these elements into uranyl silicates in order to estimate the solubility limits for Pu and Np due to these phases.

- I can find only two published measurements for the silica-saturated dissolution rate of SRL-202 glass, the production glass at Savannah River Laboratory (SRL). These measurements were on simulated waste glass. We need more testing in this area, in particular of real production glasses. Some of the hot glasses produced at SRL before the defense waste processing facility (DWPF) began operation contained significant concentrations of elements such as Cl and P, which were not reported in the composition specifications. These glasses behaved differently (more poorly) in durability tests. We should find out whether these types of glasses currently are being produced; if so, they should be tested.

- Evaporation of J-13 water in a dripping environment will produce a concentrated brine. The brine will be at a high pH if CO_2 is thermally driven off and replaced by water vapor, as predicted by some of the thermohydrologic modeling. If this is the case, the spent fuel will contact a high pH saline brine (pH 11 or higher). Even if CO_2 is at atmospheric levels, the pH will exceed 10. More data are needed on spent fuel reacting in such brines.

- As mentioned above, actinides from dissolving spent fuel could sorb onto colloidal clay particles generated during glass alteration. Corrosion tests should be performed on both glass and spent fuel to evaluate this possibility. In addition, the can-in-can glass canisters, which contain the Pu-ceramic, almost certainly will be sites for this scenario, in which actinides from another waste form contact the colloidal clay particles from DWPF glass. Some tests to evaluate this scenario should be performed.

- Little work has been done to couple the materials degradation processes. Fluid packets will be involved in a variety of corrosion processes as materials travel through the repository. Most of the work to date has focused on one material in existing groundwater chemistries. Additional work should focus on the most potentially harmful coupled interactions. For example, water that has a very high pH due to cement alteration contacting glass will have a much faster reaction rate. Both modeling and experimental studies should be used to address this issue.

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environment there will be more rapid O_2 diffusion and more oxidation, but discontinuous water films will slow subsequent release.

Modeling of the intrinsic dissolution of the waste form must consider not only the forward reaction rate of matrix dissolution, but also the potential for radionuclides to be stored in alteration products and the potential for episodic release of these products from sudden influx of water, a scenario that might lead to release of a pulse of radionuclides.

CLADDING

Degradation modes for Zircaloy cladding can be separated into pre-container failures, including current dry storage over decades, and post-container failures. Pre-container failures include the possibility of creep rupture, stress corrosion cracking, and delayed hydrogen cracking. Creep ruptures at temperatures less than 300°C are improbable, making this a highly unlikely failure mode for low- to moderate-burnup fuel. This conclusion can be supported by the valuable data developed from dry storage monitoring (Stewart et al., 1996). Initial monitoring began in 1985 and includes data from 4 casks and thousands of rods. No failures had been observed as of 1996. Low strain failures are a remote possibility. Chung et al. (1987) ascribed these to thick oxides (50-70 μm) causing increased stresses and initiating cracks at the outer wall. Such cracks did not occur for 20-30 μm films. Therefore, this mechanism is relevant only to high burnup fuel and probably is not relevant to post-container failure oxidation rates except for premature container failures. Also, the internal steel components will probably oxidize more rapidly and would consume O_2 because a defect in a container that could escape detection would be small (<0.5 mm). Yaggee et al. (1980) also reports low strain failure (0.01%), which may represent stress corrosion cracking during in-reactor ramping.

Stress corrosion cracking (SCC) will not contribute significantly to failure. It should occur only for rods with high burnup and incipient cracks and there are small numbers of such rods. Cladding failure due to delayed hydrogen cracking (DHC) is extremely

unlikely, because the stress intensities are well below the failure threshold, even assuming significant crack lengths. However, before accepting an argument that DHC will not occur, one should look at the potential for precipitation of hydrides in highly stressed regions during slow cooling. In addition, the monitored dry storage data should provide pertinent information.

To consider the potential for post-container failure of the cladding, one must consider possible conditions inside the failed canister. If the first penetrations of the canister wall are few and small, there may be a period of low oxygen availability. However, with the development of additional and more extensive canister penetrations, O_2 will become generally available, and conditions will be oxidizing. Whether drips occur is also important. In the absence of drips, the temperature and relative humidity outside the canister at the time of canister failure will quickly equilibrate inside as well. With drips, the internal environment may involve steam or advective flow, and perhaps ponding of water within the package. It is possible that the incoming J-13 groundwater will become concentrated due to evaporation and there will be a slow buildup of salts in the residual water. In addition, it is expected that massive iron oxide sludge will accumulate around the assemblies and between the rods.

Several cladding failure modes are possible given the waste package environment described above. General corrosion (oxidation) of the cladding will be essentially negligible under dry, moist, or wet conditions at temperatures below ~250°C. Because of the extremely low rates of general corrosion, failure by this mode is not conceivable. Localized corrosion (pitting and crevice corrosion) at temperatures <100°C is difficult to assess. There is no relevant database for irradiated Zircaloy, and there is no information near 100°C in dilute salt solutions, such as might develop under dripping conditions. Acidity and sufficient Cl^- concentrations, along with sufficient temperatures, are required to induce localized corrosion. Acidity can develop in occluded sites, such as at the contacts of the cladding with spacers, and could result from hydrolysis of ferric chloride. Concentration of groundwater through evaporation is a mechanism for increasing the Cl^-

concentration. However, after a few thousand years, the relative humidity (RH) in the drift will be >80% and temperatures will be <100°C. With the evolution to higher RH and lower temperatures, evaporation will decline. Therefore, there is probably a limited time during which conditions would be conducive to the development of saline conditions.

There appears to be no relevant set of studies to rule out pitting and crevice corrosion in Zircaloy. It is possible, though unlikely, that it would occur in low pH saline groundwater. The probability would be even lower under conditions of high [Cl^-] and neutral pH. Once initiation occurs, penetration is likely to be rapid. A good mechanistic study of localized corrosion of Zircaloy is needed to make a strong argument for cladding credit.

Unzipping of the cladding from oxidation of UO_2 to U_3O_8 appears to be very unlikely. Preliminary assessments, reported by E. Siegmann at WFEE Workshop 2, indicate that this failure mode is relevant only to containers that fail prematurely and in which some rods failed prior to emplacement. I suggest that to confirm this, the calculations should be reassessed using an activation energy of 146 ±10 kJ/mol (McEachern et al., 1997).

Potential secondary cladding failure mechanisms include hydriding at sites remote from initial defects, a process that could lead to hydrogen embrittlement. Long-term uptake of hydrogen in an oxidizing atmosphere will occur but will be limited by the oxidation process. Hydrides that form may undergo microstructural orientation of hydride platelets and the Zircaloy will develop poor mechanical properties. Therefore, evaluation of this process should include a consideration of the potential impact of mechanical loads that the cladding might experience after it becomes embrittled, as well as internal pressures due to swelling from oxidation of UO_2 to U(VI) oxides. Likewise, sites remote from the defects or locations of cladding failure may undergo embrittlement.

SURFACE AREA EXPOSED AND WATER CONTACT

The amount and timing of the exposure of fuel surface area is related to the nature of cladding failures, which could be cracks and/or local corrosion. Cracking failures from delayed hydrogen cracking or creep rupture result in exceptionally small failed surface areas. For example, assuming a 10 mm x 0.5 mm fracture and a total rod surface area of 0.13 m^2, the fraction of the surface area failed equals 4×10^{-5}. The surface area of failure sites due to crevice or pitting corrosion is difficult to estimate, but could be larger depending on size and number of sites. However, they would still be small compared to the rod surface areas.

The effect of the low cladding-failure surface area on fuel dissolution/alteration depends on environmental conditions. Given dry air oxidation,the gas diffusion coefficient will be high, and the defect size will have a small effect on the extent and rate of oxidation. Under saturated conditions, the rate of O_2 ingress could be greatly reduced. Reexamination of the experiments of Wilson (1987) for hole and slit defects suggests that the results can be explained by outward diffusion of U, and the fact that U is oxidizing 100 times faster than it is being released. The implications of the mass balance of O_2 and U from the experiments is that the large volume increase of UO_2 to UO_3 is filling fractures and reducing O_2 mass transport. Caveats to these conclusions are that swelling may occur, which could lead to rupture at the defect location. Also, secondary defects from hydriding could occur at sites remote from the defect; thus these regions could have high H_2/O_2 ratios. Conditions would be the most aggressive in a steam environment. There would be more rapid O_2 diffusion and more oxidation, and a film of water would contact the fuel. However, because of a lack of a continuous water film, mass transport out would be slow.

At present , it is difficult to estimate how long the limits to mass transport out of the clad fuel (dicussed above) will last or how the delays in release should be incorporated into waste form degradation and release models. Considering the small void space in a fuel

rod and the large increase in molar volume in going from UO_2 to $UO_3 \cdot XH_2O$, the reaction rate is likely to slow considerably, but in the long run, fuel rod swelling and rupture are likely.

Assuming that the relative humidities at the time the waste package and cladding fail will be high (with the possible exception of juvenile failures), essentially all of the fuel surface area will be wetted with at least a thin, discontinuous adsorbed water layer. The surface area that is used in the intrinsic dissolution model is related to how quickly the fuel matrix alters (not to subsequent transport and accumulation, discussed below). The flow-through dissolution tests imply that the surface area is the fragment area plus the boundaries of the outer few grains (Gray et al., 1993, 1994). However, the drip tests of J. Bates and his group (presented at WFEE Workshop 2) suggest that all of the grain boundaries are being attacked. It may be oxidation and not dissolution that occurs at the grain boundaries. Neither the flow-through tests nor the drip tests give releases that imply that all grain boundaries were involved. There are suggestions that an adherent alteration layer will develop that may inhibit dissolution. My assessment of the applicable surface area for the matrix dissolution model is three to five times the fuel fragment surface area.

INTRINSIC DISSOLUTION

I think that the intrinsic dissolution (forward reaction) model can be simplified so that the primary dependency of dissolution rate is temperature. Temperature, which is time-dependent, ranges from 100°C to 25°C. It is not clear whether it is necessary to carry all of the other parameters as independent variables. I suggest that criteria be established for including data in the pool, acknowledging that the data have been gathered under a range of conditions. These criteria are the set of conditions defined by the range of probable conditions in the repository. For example, the O_2 concentration should be atmospheric; all variations in CO_3 should be included; pH may range from 7 to10.5; and burnup can include both UO_2 and spent fuel. Once the data have been selected using these criteria, I

suggest that a regression be developed that is a function of temperature only. After that, the individual effect of variations in individual parameters can be assessed relative to the regression relation.

Potentially, the influence of Ca and Si could be included in modeling fuel dissolution. It is apparent from a variety of flow-through tests that these cations have an immediate effect of reducing the dissolution rate by about 100 (Tait and Luht, 1997). It is not apparent that there is sufficient Ca and Si concentrations in the expected water chemistries to achieve this effect. Furthermore, the likelihood of significant alteration/oxidation occurring in an unsaturated steam environment suggests it would be nonconservative to incorporate this in the release model because the adsorbed water film may not contain groundwater ions.

The intrinsic dissolution rate defines the alteration rate, but is an upper bound on the radionuclide availability rate, because some radionuclides are solubility-limited and because alteration to secondary phases may retard radionuclide release.

RAPID RELEASE FRACTION

As reported in my talk at WFEE Workshop 2, a study was conducted (Johnson and Tait, 1997) with the objective of estimating the instant release fractions for key radionuclides for performance assessment calculations. The approach taken was to estimate the average fission gas release for Swedish boiling water reactor (BWR) fuel and to correlate gas release with fission product leaching data from various studies. I suggest that these estimates are reasonable to use for this project because the leaching data is also based partly on studies of American light water reactor (LWR) fuel. The instant release fractions are conservative in that they include both the gap and grain boundary inventories.

STABILITY OF SECONDARY PHASES

Assuming that the intrinsic dissolution rate provides the rate of alteration of the fuel, the portion of the radionuclides that are available for transport should be reduced by the fraction that precipitate in secondary phases and that are absorbed to these phases. Apparently, the previously elicited values of radionuclide solubilities were based largely on thermodynamic assessments, thus implying concentrations that are far higher than the observed aqueous film concentrations from the recent unsaturated spent fuel tests. Because the TSPA needs a mobilization rate that accounts for retardation due to secondary phases as well as colloids, the previously elicited solubilities appear to be very conservative. The report by Sassani and Siegmann (1998) illustrates that spent fuel dissolution experiments give Np concentrations that are reasonably consistent within these types of experiments but are far lower than those from pure-phase solubility studies or thermodynamic calculations. Given that conditions within the waste package will give rise to production of voluminous quantities of U(VI) alteration products, the range of Np concentrations from spent fuel experiments appears to be directly relevant and should be used. Allowing for a maximum value that is several orders of magnitude higher, as given in Fig. 3.1 of Sassani and Siegmann (1998) seems sufficiently conservative. I do not see the relevance of the pure phase solubility studies in defining this distribution, because it is not clear how they could form if low concentrations of Np are established and maintained by complex mixed U/Np phases. If the performance assessment includes some bounding value/worst case calculation as part of the suite of calculations, perhaps these pure phase solubility values could be retained for such an application.

In modeling the mobilization of solubility-limited radionuclides, it is prudent to consider a scenario in which there is alteration and accumulation of nuclides in alteration products, followed by a pulse of water that washes out the alteration products. This may occur, for example, because of climate change or from the change from a steam environment to a dripping environment in which there is significantly more advective flow.

COLLOIDS

Tests show that a significant fraction of the Am and Pu is colloidal (Wilson, 1988). Nevertheless, the real issue is how colloids flocculate, the mass transfer that occurs, etc. It is conservative to assume that the colloidal production rate equals the dissolution rate, and then consider the mobilization and transport properties of the colloids (e.g., how do they get out of the waste package?). The potential for colloid release from the waste package is determined far more by the local flow conditions than by the specifics of the alteration phases or the alteration process itself.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

A good case can be made for cladding credit only if it can be shown that localized corrosion of Zircaloy will not occur under the range of conditions expected within a failed waste package. Because this appears to be the only mechanism that might lead to failures, it would be very valuable to perform studies of the initiation of local corrosion of Zircaloy.

The most probable exposure mode for the spent fuel appears to involve a steam environment (adsorbed water film), for which few data are available, and none at all at temperatures near 100°C. The alteration rate should be measured under such conditions.

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ELICITATION SUMMARY
HENRY F. SHAW

OVERVIEW OF WASTE FORM DEGRADATION PROCESSES

Once a waste package has been breached, important issues arise related to the reaction of air (oxygen), water vapor, and liquid water with the waste forms (including the cladding of spent fuel). Whether the cladding has failed and how it fails obviously will impact the timing and means of exposure of the fuel and consequent releases, though I do not believe that credit can be taken for the cladding. Even in the absence of liquid water, the waste can undergo changes due to hydration by water vapor (both glass and fuel) or oxidation air (fuel). Both the form (vapor or liquid) and quantity of the water that contacts the waste form are significant. Characteristics of the fuel, including fission gas and possibly burnup (in addition to the obvious relationship between burnup and inventory), should be considered. The intrinsic dissolution rates of the waste forms, as well as the development of secondary phases and their solubilities as a function of the evolving chemical system, are important issues. Colloids entering and generated within the waste package may be important to the mobilization of radionuclides, depending on how effectively they can be transported out of the package.

It is important to consider all possible environmental scenarios. For example, the contents of some packages may be exposed to relatively large quantities of liquid water (those near dripping fractures), while others may interact only with water vapor. The set of individual packages that is "wet" at any instant will change over time as the thermo-hydrologic system evolves and old fractures seal and new ones form. Furthermore, because of the possibility of wet periods, in which a larger number of packages may be exposed to dripping water, it will be difficult to argue that the total number of wetted packages is constant over time. This situation poses a difficult modeling problem, because the instantaneous radionuclide release behavior of a given waste package is not just a

function of the state of the system at that moment. Rather, the packages have "memory," and the release behavior of a waste package depends on its entire history up to that time. For instance, fuel that was never exposed to liquid water, but that had been oxidized and hydrated due to exposure to air and water vapor, will have different release characteristics when it is finally contacted by liquid water than would fuel that is still substantially UO_2 when it is first contacted by water. Similarly, waste glass that is alternately wetted and dried will have a different release behavior than one that is completely vapor-hydrated before it is contacted by liquid water. In conducting a performance assessment, it will not be sufficient to consider X percent of the ensemble of packages as wetted by dripping water, Y percent as being "bathtubs", and Z percent as in contact with the humid, ambient air. Packages will transition between states, and a package's behavior in state X will be different if it arrived in that state after passing through state Y than it would be if it arrived there after passing through Z. Some means must be found to account for these histories when considering the entire ensemble of packages.

An early-breached waste package could, if the penetrations are limited to small openings, have lower-than-ambient oxygen partial pressures within the package if the consumption of oxygen by oxidation of the cladding and/or UO_2 fuel exceeds the rate of supply of oxygen through the penetrations. The balance between the consumption and supply rates will determine the actual oxygen partial pressure during this time. With time, however, as oxidation rates decrease due to falling temperatures, and the penetrations enlarge or increase in number, the potential supply rate of oxygen to the interior will exceed the consumption rate, and the internal atmosphere will approach the composition of the atmosphere outside the package. Given the uncertainties in the types, sizes, and number of penetrations that will occur in a failed waste package, and the uncertainties in the rates of the processes responsible for the consumption and supply of oxygen, it does not seem likely that the Project will be able to take credit for an early period of reducing conditions within failed waste packages. It will be necessary to assume that the atmosphere in a failed package is the same as the surroundings.

It is likely that aqueous fluids inside failed waste packages will have higher concentrations of solutes than the incoming fluid, even in the absence of significant dissolution of waste or waste package materials. This is because the waste package will always be warmer than the wall rock and the relative humidity will be lower inside than outside, leading to evaporation of water. This may affect the corrosion rates of metals and possibly the dissolution rate of glass and fuel.

CLADDING

Spent-fuel cladding is not my primary area of expertise, but I will offer the following observations.

Although it is physically reasonable to take credit for cladding, it is my opinion that there has been insufficient work in this area to justify including the cladding as a key barrier to the release of radionuclides from spent fuel. Moreover, I am not sure what work would suffice to be able to provide the technical basis for including a cladding barrier in the license safety case in any manner other than a redundant, and "extra credit" barrier. In my opinion, the uncertainties associated with the behavior of the cladding are, and always will be, irreducibly large.

Two roles have been ascribed to cladding credit. Gradual failure of the cladding would result in a distribution over time of the onset of release of radionuclides from the spent-fuel pellets. This effect would act to smear any pulsed releases, especially of highly soluble nuclides in the gap, or rapid release, inventory. Secondly, if the cladding fails through a small number of small (pinhole) through-going penetrations, the cladding could act as a transport barrier to both advective and diffusive transport of radionuclides. I do not think that a quantitative credit can be assigned and defended for either role, and certainly not based on the current state of understanding.

If one requires that the cladding fail gradually, then one must be able to rule out that cladding failure can be induced by some environmental perturbation that affects many packages. For instance, if localized corrosion by concentrated solutions is found to be a viable failure mechanism, then it is possible that the environmental conditions necessary for this to occur could develop nearly simultaneously for a significant fraction of the waste packages. That is, one must be able to rule out common failure modes. An example of this might be a pulse of water during a wet period that washes accumulated salts from either the host rock or the surface of the waste package into the contact with cladding. To rule this out, one must have a clear understanding of both the likely cladding failure mechanisms and possible environmental changes. Turning to the second desired role of the cladding, if one wishes to take credit for the cladding as a transport barrier, then one must be able to predict with confidence the distribution of the number, aperture sizes, and locations of the cladding penetrations as a function of time. The locations are important because they provide information on the length of possible diffusive paths and on the likelihood of the occurrence of an advective path. The penetration dimensions are important because they likewise are needed to calculate the impedance they represent to both advective and diffusive transport. Finally, the number of penetrations is important because the total flux out of a failed pin is, to first order (in the absence of interactions between penetrations) the sum of the fluxes through each penetration. In addition, it can be shown that only a relatively small number of penetrations are required before the cladding no longer acts as an effective barrier to diffusive transport. In short, I do not think that defensible estimates of these parameters can be established.

There are a number of possible failure mechanisms for cladding. The statistics on in-reactor failures appear to be fairly well constrained. There may also be some failures associated with the transportation, handling, and loading of spent fuel into waste packages. Within the waste package, generalized corrosion will occur upon exposure to oxygen, but this process will be extremely slow, and I agree that it is unlikely that failures due to this mechanism will occur on any timescale of interest. Localized corrosion is not likely in dilute water, but because of the potential for concentrated solutions formed by alternating wet and dry periods, localized corrosion may be an important failure

mechanism. In addition, there will be nooks and crannies in the fuel assemblies (e.g., where the pins rest on spacer plates) that form crevices that could attract the last bit of water before it evaporates. It seems quite possible that localized corrosion could occur in such locations. Hydride reorientation remains plausible to me, because there will be temperature gradients along the pins, which will lead to transport of hydrogen. Unlike most other failure mechanisms for the cladding, this mechanism can occur even in the absence of a waste package failure. Finally, over the long term, mechanical failures of the cladding are likely due to processes such as collapse of the waste package and/or rockfalls.

The project should examine the reasons for the desire to take credit for the cladding, and try to find some other means of providing an equivalent reduction in the source term. Preferably, this reduction would come from consideration of an engineered barrier, not one that is as poorly characterized and difficult to model as the cladding. Clearly, the most defensible position is to take no credit for cladding in the primary safety arguments, and to argue that the presence of cladding can only act to reduce radionuclide releases by a poorly quantified, but potentially large, amount.

SURFACE AREA EXPOSED AND WATER CONTACT

A small fraction of waste packages will have undetected failures at emplacement, and some packages may suffer mechanical failure (e.g., rockfall) during the period of above-boiling temperatures. The contents of these packages will be exposed to the ambient atmosphere at relatively high temperatures (above ~95°C). Most waste packages, however, will fail after the drifts have cooled to below the boiling point of water. Even under such conditions, when a waste package first fails, it is likely that the contents (i.e., the waste form) will not initially be contacted by large quantities of liquid water. Instead, the interior of the package will be exposed to the ambient atmosphere, which will be close to saturation with respect to water. Experimental evidence (from the vapor hydration tests at Argonne National Laboratory [ANL]) shows that both spent fuel and

borosilicate glass are hygroscopic, and a thin film of water will form on the exposed surfaces of these materials. Thus, the waste form surface must be treated as being completely wetted by this liquid water film, which can and will react with the material. The concentration of solutes in this film will be quite high, owing to the small volume of fluid relative to the surface area with which it is in contact.

Although the waste form surfaces will be wetted, there may not be transport of radionuclides from the surface. Transport out of the package requires either a continuous fluid pathway (for diffusive transport), or an advective path, which most likely would involve dripping or an overflowing "tub" scenario. Note that there appears to be evidence (again from the ANL vapor hydration tests) that sufficient water can condense on the surface of spent fuel exposed to water-saturated air to cause liquid water to drip off the surface. These drips transport radionuclides from the surface of the waste form in these tests, and may do the same under repository conditions.

The release rate of solubility-limited nuclides will not depend on the wetted or exposed surface area. These releases depend only on the advective flux in the case of advective transport, or the diffusional path length and concentration boundary conditions for the case of diffusional transport. The wetted area only figures into the calculation of the release rate of highly soluble nuclides, including those present in the rapid release inventory. When considering the release of these elements, one needs some estimate of the reactive surface area of the waste form.

There are conflicting data and diverging lines of argument over how the reactive surface area of spent fuel will evolve as it degrades. If the fuel is exposed to humid air at temperatures >120°C, there is little question that preferential grain-boundary oxidation will occur, and the surface area available for later contact by liquid water is that of the individual grains. However, the situation is not as clear under other conditions. The data presented and published on the unsaturated (drip) tests at ~90°C on spent fuel at ANL seem to show no preferential dissolution or hydration along grain boundaries. Nevertheless, the fuel fragments in those tests fall apart, apparently into individual grains,

when subjected to slight shear forces, implying that the grain boundaries have been extensively weakened, possibly by preferential grain-boundary oxidation (but not hydration). J. Bates (WFEE Workshop 3) said that fuel fragments from longer duration tests now show evidence of possible enhanced dissolution or hydration along grain boundaries, and that he is no longer sure of the previous ANL conclusions, but the supporting data were not presented. In hydraulically saturated dissolution tests (C. Wilson's tests at Westinghouse Hanford Co., now Pacific Northwest National Laboratory [PNNL]) there was evidence of preferential grain boundary dissolution; upon disassembly, the tests were found to contain significant quantities of individual fuel grains that were not present when the tests began. Thus, there appears to be good evidence that the surface area available for contact by water will evolve quickly from the surface area of fuel fragments toward that of individual grains.

Countering this view is the fact that in the unsaturated and vapor hydration tests of fuel, secondary alteration products form on the surface of the fragments. These minerals, which have much larger molar volumes than the parent UO_2, coat the surface of the fragments and might ultimately fill the inter-fragment void space. If such minerals form along grain boundaries (if the boundaries are subject to preferential attack), then the grain boundaries might actually be self-sealing. R. Stout (WFEE Workshop 2) argued that these processes would therefore reduce the available surface area of spent fuel over time, and that the use of the geometric surface area of the fuel fragments provides a conservative estimate of the accessible surface area for all time. Clearly, this is the less conservative case, and I do not think we have sufficient data to support the use of fragment area as a conservative case.

RAPID RELEASE FRACTION

Rapid release from the fuel/cladding gap and grain boundaries is important for a number of radionuclides, including Cs, I, and ^{14}C. The study conducted by the Atomic Energy of Canada Limited (AECL) for the Swedish Nuclear Fuel and Waste Management Company

(Johnson and Tait, 1997) brings together release data on a variety of radionuclides that might, a priori, be considered as candidates for a rapid release fraction. The available data (of which there is not a superabundance) appear to indicate that the fractional fission gas release provides a reasonably defensible upper bound on fractional gap and grain boundary inventory, and that only a few elements (notably Cs and I) even approach this limit.

To assist the performance assessment, however, it will be necessary to have a defensible projection of the distribution of fission gas releases in both the existing stock of spent fuel and all future fuel destined for a Yucca Mountain repository. The uncertainty in those projections may well dominate the uncertainty in the available rapid release fraction.

INTRINSIC DISSOLUTION RATES

Like surface area, the forward dissolution rate is important only for non-solubility controlled elements, such as Tc and I, and colloidally transported elements. For TSPA-VA, a fully mechanistic, scientifically rigorous dissolution rate model is not needed. What is needed is a rate expression that:

- recovers available experimental and analog data to within a known uncertainty,
- is computationally tractable,
- depends only on known or predictable environmental and waste form-related variables, and
- is valid over the range of variables accessed during the TSPA simulation.

With some reservations, the empirical dissolution-rate expression for spent fuel that was proposed by the Yucca Mountain project team may satisfy these requirements. There are no obvious trends in the residuals. The regression model recovers the YMP flow-through and drip-test data to within a factor of two or three, which is as good as the inter-laboratory reproducibility for these tests. The independent variables in the model are

knowable, and the mathematical expression used is relatively simple. However, other experimental data were shown (a compilation of data from a variety of sources that was presented by J. Tait, WFEE Workshop 2) that show factors of 10 to 100 differences in the dissolution rates for nominally the same sample types and test conditions. These unexplained differences are much larger than the changes in dissolution rate due to dependencies on variables such as $P(O_2)$, temperature, or burnup, at least over the range of relevant conditions. I suspect, but have no proof, that a portion of the unexplained differences could reflect systematic differences in the manner in which the surface area of the test specimens is calculated, measured, or estimated. This is unlikely to account for the entire factor of 100, however. These differences, until explained, imply quite large confidence limits on any prediction of dissolution rates for spent fuel.

My reservations with respect to the empirical regression model are outlined in the following paragraphs.

- The regression has not been tested by attempting to predict the results of tests that were not used in the regression (other data from AECL, Spain, etc.). Doing this would provide a better estimate of the true uncertainties in the predictions. In conversations with Ray Stout (WFEE MDT member), it appears that there are actually few, if any, published data that are suitable for this test; most publications omit key information. A concerted effort should be made to obtain the information needed to conduct such an analysis.

  An empirical expression may not be sufficient for licensing if the terms in the expression (for example, the cross-terms) do not have some mechanistic basis or rationale for their functional form.

- The expression clearly is not valid over the range of possible conditions. For example, the log(variable) terms introduce singularities at zero values of variables. In some cases this is because there is a change in mechanism (*e.g.*, we know that there is a change in mechanism at low $P(O_2)$, and that the rates of oxidative dissolution reach

zero). However, no such explanation is available for the log (burnup) term. Indeed, in conducting the regression analysis, the burnup of unirradiated UO_2 was arbitrarily set to 1MWD/KgHM because a nonzero value is necessary due to the functional form.

- There are no data for low pH and few data at high burnup, thus the functional dependence of the dissolution rate on these parameters is poorly constrained. Data are needed on higher/lower burnup fuels and over a wider range of pH to show if the putative dependences are real.

- The dependency on burnup, if real, must reflect "lumped" effects of fuel chemical composition (fission products and transuranic actinides), gamma and alpha radiolysis, and possibly other factors. Some of these parameters have time-dependent values, and it is not clear that a regression using young spent fuel would be valid for fuel that has undergone several thousand years of radioactive decay. For instance, if the burnup effect is real, and is due to the gamma radiation field, the apparent dependence on burnup will change over time.

- The model does not account for solution species other than O_2, CO_3^{2-}, H^+, and (possibly, through the burnup dependence) radiolytic oxidants.

- The errors in the regressed data should be formally propagated (i.e. the data regression should used weighted observations).

We were asked whether, if the empirical rate expression is used, it would be better to drop some of the independent variables from the regression, or hold them constant as "average" values (which may be a problem for non-linear terms). I suggest that the existing regression fit be used, and that the effect of fixing some independent variables be evaluated by sensitivity analyses.

The glass dissolution model is a semi-empirical model. Its shortcomings were well covered in the presentation by W. Bourcier at WFEE Workshop 1. The long-term rate has a poor theoretical basis, and it is probably temperature dependent. The model does not account for dissolved species other than H_4SiO_4 and H^+. It may be overly conservative inasmuch as it does not account for precipitation of radionuclide-containing secondary phases.

STABILITY OF SECONDARY PHASES

From the tests conducted to date, I am comfortable that the secondary phases that form in these tests are the same ones that are likely to form by alteration of the fuel in a repository. My only caveat is if alloys or other materials are used in the engineered barrier system that contain elements that have not been considered in the waste form alteration tests (e.g., vanadium or copper.).

I also believe that the release of actinides will be controlled by the solubility of actinide-bearing secondary phases, even when the element in question is not present in high enough abundance to stabilize it own phase. In such cases, the low-abundance actinide will be present in solid solution in other actinide phases. For example, Np has been detected in secondary uranyl phases in work at ANL. Despite my personal beliefs, there are not sufficient data to quantify this process or to provide assurance in a safety case that this process can be counted on to limit releases. An experimental program should be launched immediately to provide supporting data. Ideally, one would experimentally determine a water-mineral partition coefficient for each element that does not form its own phase. Such data would have to be collected for each phase and for a range of water chemistries. Combining that knowledge with knowledge of the secondary phases that form would allow one to calculate the concentration of those low-abundance elements in a coexisting solution. To be able to make this argument, one needs to identify the important secondary phases and show that the radionuclides are in solid solution in the secondary minerals, and that the measured partitioning is reversible and represents an

equilibrium situation. Experimentally determined partition coefficients would also provide a partial validation of the analysis of the low-drip-rate tests (see below).

I do not think that the effective solubilities derived from the low-drip tests results should be used in TSPA-VA. They might be used in a design analysis (an analysis of the most likely releases) but not in a safety analysis (an analysis in which one purposely assumes the worst and shows that the consequences are acceptable). The model being used to interpret the results has not been validated, and has been applied to only a small data set at only one drip rate. Simplifications in the analysis as presently conducted means that the model cannot be used to interpret the high-drip-rate data. Additional work is necessary to validate the analysis, such as a new drip-test at drip rates between those of the vapor hydration tests and the low drip-rate test. The release rates implied by the drip tests could be included in the PA as a sensitivity test to see how important secondary phase stability is.

After the UO_2 is completely reacted (to hydrous uranyl oxides and silicates), continued release will be controlled by the dissolution rate of these secondary phases. Unlike UO_2 these phases have the potential for being in thermodynamic equilibrium in the repository, and when at equilibrium, would have a net dissolution rate of zero. Because of the fine-grained nature of the alteration products, they will have very high surface area, and any perturbation away from equilibrium (e.g., by ingress of clean water) will be quickly erased by the dissolution (or precipitation) of the secondary phases. By definition, all high-solubility (i.e., non-solubility-limited) nuclides will have been flushed from the waste package. All further releases therefore would be dependent only on the advective flux and solubility limits. The releases of other radionuclides incorporated in these phases (minor or low-abundance elements such as Np) will be controlled by the dissolution rate of the secondary phase, and one can safely assume a congruent release of the incorporated nuclides.

It should be noted that assuming high solubility limits for secondary phases is not a conservative assumption when considering the possibility of nuclear criticality within

degraded waste packages. In this case, one is concerned with the accumulation of fissionable nuclides (e.g., Pu) in secondary alteration products. Setting a conservatively high solubility for these elements in order to maximize releases will result in a non-conservative estimate of the retention of these elements in the waste package.

Np SOLUBILITY LIMITS

There has been considerable interest in the solubility limits of Np, because recent performance assessments that used a high solubility limit for this element predicted that releases of Np to the accessible environment would be large, with accompanying large doses. These high solubility limits were arrived at through an expert elicitation, and were motivated in part by some experimental work (by H. Nitsche et al., 1993, 1994) that suggested the thermodynamically stable phase (NpO_2) has quite high solubility. The high values were considered despite the fact that experimental results on actual waste forms suggest much lower limits to the Np concentration in solution. Nitzche's data have been called into question, and a position paper by D. Sassani and E. Siegmann (1998) argues that the experiments were metastably supersaturated. Sassani (ibid, and presentation at WFEE Workshop 2) argues that the experiments attempted to approach equilibrium from oversaturation, but that kinetic inhibitions to nucleation and precipitation of the equilibrium Np-bearing phase prevented it from forming. He further argues that such conditions would not be reached in a repository, because Np saturation would be approached from the undersaturated side. This argument does not hold up to scrutiny. If there are kinetic barriers to precipitation of the stable phase, it does not matter from which direction one tries to approach the equilibrium.

Sassani put forward a second line of argument for using a lower solubility limit for Np in PA. He invokes the possibility that the Np is being incorporated in solid solution in secondary alteration phases of spent fuel, and that apparent low solubility in solutions in contact with fuel is a result of a large solid/fluid partition coefficient for Np. I find this a

much more plausible argument; however, as discussed above, the experimental data needed to validate this idea have not been gathered.

Finally, I am aware of some recent results by Tom Wolery and Cindy Palmer of Lawrence Livermore National Laboratory (personal communication) that show that NpO_2 does precipitate from supersaturated solutions at ~200°C, and that the coexisting solutions have Np concentrations several orders of magnitude lower than were assumed for solubility limits in recent performance assessments. Oddly, this work was not funded by the Yucca Mountain project. I strongly recommend that the project undertake to obtain experimental data to settle questions such as these, rather than continuing to attempt to resolve them by voting.

COLLOIDS

Colloids are operationally defined as particles that can be filtered out of solution but do not settle. Colloids fall into two categories, both of which can affect radionuclide transport. The colloids may be composed of polymerized forms of a radionuclide (so-called true colloids, of which Pu colloids are an example). Alternatively, the colloids can be non-radioactive particles such as mineral fragments from the surrounding rock, or alteration products of waste package materials (so-called pseudocolloids). There will be numerous sources of colloids in the repository, including Pu released by degradation of the waste forms, zeolite and clay particles in the groundwater, and alteration products of waste package materials such as ferric hydroxides. The alteration of the HLW glass will also generate clay and zeolite colloids. The colloids observed to have been mobilized from the surface of the waste form in the drip tests conducted at ANL could have been transported by convective processes or knocked off the surface by mechanical shock. However, similar processes also exist in nature (drips, rockfall, earthquakes, *etc.*) and likewise could mobilize colloidal particles within the waste packages. There is no question that colloids exist, and that they can carry radionuclides. Indeed, the issue of

colloids is not really one related to waste forms. Instead, the key question is how effectively they can be transported out of the package and through the unsaturated zone.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

Recommendations are given in the relevant sections above.

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ELICITATION SUMMARY
DAVID W. SHOESMITH

OVERVIEW OF WASTE FORM DEGRADATION PROCESSES

The overall degradation process of the waste form and the mobilization of radionuclides is affected by a sequence of events that comprises cladding degradation, dissolution of the instant release fraction, waste form dissolution and the release of radionuclides incorporated in the fuel matrix, and the influence of secondary phase formation on fuel dissolution and retention of radionuclides.

Various failure mechanisms are possible for fuel cladding, including some that are independent of repository environment (creep failures and those due to stress corrosion cracking [SCC] and/or hydrogen-induced cracking [HIC]) and some that could occur after failure of the waste package (due to embrittlement or general and/or localized corrosion).

Immediately after cladding failure, depending on the time required for water to gain access to the fuel waste form, rapid release of the cladding gap inventory and the grain boundary inventory (predominantly ^{137}Cs and ^{129}I) will occur. The amount released will be independent of the corrosion properties of the fuel, but rather determined by its in-reactor history. Subsequent radionuclide releases will be determined by the corrosion behavior of the fuel and will depend on the environmental conditions established inside the waste package after failure and on the ingress of water. Preoxidation of the fuel in humid vapor prior to exposure to aqueous conditions could accelerate fuel dissolution by loosening grain boundaries, thereby increasing the surface area available for dissolution.

The environment within the waste package will help determine fuel corrosion rates. In particular, temperature, oxygen availability, water chemistry, and the presence or absence

of seepage (drips) will be important. For both cladding failure and fuel dissolution, the composition of the water to which they are exposed will be very important.

Eventually, as a consequence of the fuel corrosion process, deposits of corrosion product will form on the fuel surface. These deposits may affect the rate of radionuclide release in a number of ways: (1) they could increase the rate by inducing cladding rupture, leading to an increase in fuel surface area exposed; (2) they could decrease the rate by blocking the fuel dissolution process; and (3) they could decrease the rate by immobilizing radionuclides (e.g., ^{237}Np, ^{231}Pa, ^{239}Pu) by co-precipitation.

CLADDING DEGRADATION

Two cladding failure scenarios follow failure of the waste package: pre-emplacement failures due to creep rupture, SCC, and HIC; and failures due to corrosion of intact cladding on exposure to the in-package environment.

For temperatures below 300°C, creep failure is improbable, as discussed in detail by E. Siegmann (WFEE Workshop 2) and L. Johnson (WFEE Workshop 3). Failures due to SCC and HIC also should not contribute significantly. Approximately 0.1 to 0.01 percent of clad assemblies will have failed (by strain failure, possibly due to in-reactor power ramping) before emplacement. These failure sites are usually small cracks. When the waste package fails, these cladding rupture sites will allow the in-package environment rapid access to the fuel.

If waste package failure occurs early while fuel assembly temperatures are high, and hence in-package relative humidities low, then air oxidation of the fuel ($UO_2 \rightarrow U_3O_8$) would be accompanied by a large expansion in fuel volume. Because the fuel oxidation product (U_3O_8) would remain in place in the absence of dissolution (and hence corrosion product transport), large stresses could occur and could lead to cladding rupture. This situation could arise only for that small fraction of fuel assemblies emplaced with a defect. Such an oxidation would also increase the fuel surface area available for

corrosion by approximately a factor of 150. Based on the temperature dependence of this fuel oxidation rate, E. Siegmann (WFEE Workshop 2) calculated that the process would be negligible if containment by the waste package exceeds 100 to 150 years. For such short times, the fraction of waste packages failed will be negligible.

If waste package failure does not occur until temperatures are too low for significant air oxidation of the fuel, then the fuel will be exposed to humid-air conditions. This condition will also lead to fuel oxidation ($UO_2 \rightarrow UO_3 \cdot 2H_2O$) involving a volume expansion, but in the presence of condensed vapor, the corrosion product will be mobile and move within the cladding. In this situation, the development of large stresses leading to cladding rupture is not expected in the short term. Indeed, the accumulation of the corrosion product within the cladding could exclude the humid-air atmosphere and significantly reduce the fuel corrosion/radionuclide release rates. It remains to be demonstrated whether such a retardation process will be significant. In the longer term, embrittled cladding could be ruptured by the accumulation of such deposits.

It is also unlikely that a significant number of the preexisting failure sites will be exposed to seepage drips. Their location will be random within the waste package, and there will be no correlation of cladding failure sites with seepage drip sites. Consequently, only a negligibly small number of pre-failed cladding sites are likely to experience drips and the enhanced fuel corrosion rates that would ensue.

If in-package cladding failure is to occur, then an environment that is aggressive to Zircaloy must develop. In the absence of seepage drips, general humid-air or aqueous corrosion would occur. At sites that experienced seepage drips, saline, acidic, oxidizing conditions could develop. For these sites, the possibility of pitting and crevice corrosion must be considered. For crevice corrosion to occur, a pre-creviced site must exist on the cladding surface. Such a site could be a contact point between the cladding and the internal structure of the waste package, for example, the stainless steel support structure.

Passive general corrosion rates of Zircaloy cladding are expected to be extremely low (D. Franklin, WFEE Workshop 2). Under oxidizing conditions, corrosion would involve formation of oxide film ($Zr + O_2 \rightarrow ZrO_2$), and Franklin's conclusions would apply. However, within the cladding it is possible that deaerated conditions could occur due to the rapid consumption of available oxygen by the fuel oxidation process. Under such conditions, corrosion of Zircaloy would proceed by reaction with water and the local formation of hydrogen ($Zr + 2H_2O \rightarrow$
$ZrO_2 + 2H_2$).

Although the oxide layer should act as a barrier to the absorption of hydrogen into the metal, insufficient evidence exists to claim that the oxide is totally impermeable, especially to hydrogen produced in the atomic state (H^o). For titanium, the absorption of hydrogen into the oxide (and eventually into the metal) occurs coincidentally with redox changes in oxide composition ($Ti^{IV}O_2 \rightarrow Ti^{III}OOH$) (Shoesmith et al., 1997a). Because such redox transitions are much less favorable in zirconium (which does not readily form the Zr^{III} state), the absorption of hydrogen into zirconium oxide should be less favorable than into titanium oxide. However, impurities, especially those segregated into intermetallic particles, can act as oxide windows for hydrogen absorption into such metals. For example, grade-12 titanium, which contains segregated Ti_2Ni particles, has been observed to absorb hydrogen even under low-temperature (~25°C) oxidizing aqueous conditions (Oriani, 1987). The presence of nickel in Zr alloys also enhances hydrogen absorption (Schutz and Thomas, 1987). In view of these possibilities, and the lack of experimental evidence to exclude them, it is judicious to assume that sites exist at which the cladding can experience hydrogen absorption.

In the absence of evidence to determine the rate of such a process on Zircaloy, we can adopt the procedure recently presented to predict possible failures by the process in titanium (Shoesmith et al., 1997). Failure of the cladding as a consequence of hydrogen absorption could occur once the amount absorbed (H_A) exceeded a critical amount (H_C). The path to failure is illustrated schematically in Figure DS-1. The rate of hydrogen absorption will be a function of the passive corrosion rate. The failure criterion is

conservative, because it is assumed that failure will occur once the critical concentration is achieved. This criterion ignores the need to initiate a crack and the requirement that a sufficiently large stress be applied for crack growth (via HIC) to propagate to failure. If the cladding is subjected to a sufficient load, then failure could occur when H_C exceeds the amount required to reduce the fracture toughness; i.e., failure would occur when HIC becomes possible. For no load, failure would become possible only when H_C exceeds the embrittlement limit. For this last condition, it is likely that an impact would be required for failure to occur. Two sources of the required load are impact due to structural collapse within the waste package, and the development of even moderate stresses due to the accumulation of fuel alteration products within the cladding. Again, the location of such failure sites would not correlate with drip sites.

By comparison to our analysis for titanium, the amount of hydrogen absorbed, H_A, would be proportional to $f_H \cdot R_{FG}$, where f_H is an absorption efficiency and R_{FG} is the oxide film growth rate. For titanium, f_H was taken to be in the range of 0.1 to 0.02, where 0.1 was considered a conservative upper limit. Similar values could be adopted for Zircaloy. The film growth rate can be estimated from the data presented by D. Franklin (WFEE Workshop 2). Of major importance is that the cladding already contains hydrogen in excess of the solubility limit. In the following estimation, only the hydrogen content of the metal is considered, although it is recognized that the number, location, and alignment of hydrides also will be important.

Because I do not know the critical hydrogen contents for zirconium, I will estimate them by comparison to the values for titanium, assuming that their differences will be proportional to the differences in hydrogen solubilities for the two materials (Table DS-1).

TABLE DS-1

Property	Hydrogen Concentration ($\mu g/g^{-1}$)	
	Titanium	Zircaloy
Solubility	~10	~1
H_c (HIC)	~500	~50
H_c (Embrittlement)	~4000	~400 to 1000
As Received Content	~50	>200

Based on this simple comparison, I estimate that failure by this process could occur between 10^3 and 10^4 years after waste package failure.

For the cladding to be rendered susceptible to localized corrosion processes, such as pitting or crevice corrosion, a saline environment must develop within the failed waste package. When considering the corrosion of the C-22 inner barrier material, it was judged in the waste package expert elicitation process that a sequence of seepage wetting/drying cycles could produce an environment close to saturated in Cl^- and that the concentration of Fe^{3+} ions from carbon steel corrosion could cause suppression of pH, possibly to as low as ~2.5. It was judged that sites on the C-22 surface subjected to this chemistry would be rare and, of course, possible only on waste packages subject to seepage drips.

With regard to cladding, the question is whether such sites can form within the waste package on the cladding surface. Because the cladding remains a heat source for as long as 100,000 years, a similar sequence of wetting/drying events could lead to the concentration of salinity on the surface of the cladding. Given the "scaffolding" arrangement of the internal structure within the waste package, seepage drips are likely to trickle down this structure and find the contact sites between the cladding and the supporting stainless steel structure. These contact points constitute naturally creviced sites at which occluded chemistry can develop. The simultaneous concentration of

cations from waste package corrosion (Fe^{3+}, Ni^{2+}, Cr^{6+}, Mo^{6+}) could render these sites acidic, with a pH < 3.

A number of literature reports show that Zircaloys will pit when exposed to sufficiently saline conditions. A Cl^- concentration of 0.5 mol/L^{-1} or greater, a concentration feasible for seepage wetting/drying conditions, appears sufficient. Because O_2 and the aggressive Fe^{3+} ion are present, conditions will be oxidizing. Unlike for C-22, the repassivation potential for Zircaloy appears to be sufficiently low that localized corrosion could initiate under aerated conditions, in the absence of a significant Fe^{3+} concentration. Although I assume that the temperature threshold for pit or crevice initiation could be reasonably high, I am unaware of any data for determining that threshold.

I conclude that there is a possibility that crevice corrosion could initiate on the cladding. The probability of initiation should decrease with time, as the temperature falls and the ability to develop saline, occluded sites decreases. Once initiated, pit or crevice propagation could lead to rapid failure of the thin cladding wall, although, as far as I am aware, no evidence exists upon which to base a crevice penetration growth law. In the absence of any data upon which to base an estimate of the probability of crevice initiation (and hence cladding failure), I assume that the probability will be the same as that for the initiation of crevice corrosion on alloy C-22. Since, in the absence of any credible crevice growth law for crevice corrosion, failure would be assumed to rapidly follow initiation, the time taken to establish the aggressive environment becomes the critical feature determining when failure will occur. Such rapid failure is likely to lead only to small apertures. The number of failure sites will be the number of packages experiencing drips (N_1) times the number of occluded sites (N_2) times the probability of initiation (P). While N_1 and N_2 could be large, P will probably be small. Since the failure sites will be small, they will possess the ability to impede radionuclide transport.

In summary, I believe that, for packages not subject to seepage drips, the most likely failure process is embrittlement in approximately 10^3 to 10^4 years. Failure would occur from an impact when the internal waste package structure collapsed, or when stresses due

to fuel alteration became excessive. When seepage drips are present, crevice corrosion is possible; I assume its probability of occurrence is the same as that of localized corrosion on the C-22 inner barrier, and that failure would occur as soon as aggressive conditions were established.

RAPID RELEASE FRACTION

The rapid (instant) release fractions proposed by L. Johnson (WFEE Workshop 3), and based on a review of available data and uncertainties, should be used. Although there may be some delay in release related to the geometry of cladding failure, which may be minor cracks due to mechanical defects or pits from localized corrosion, this delay is likely to be less than the 100-year time steps used in performance assessment calculations. This fact makes release effectively instantaneous.

SURFACE AREA EXPOSED AND WATER CONTACT

One mechanism for significant volume expansion and increase in reactive surface area is preoxidation. However, as discussed above, this process is avoidable given 100 to 150 years of waste package containment. Discounting this preoxidation process, questions remain regarding the surface area of fuel that will be available for corrosion and radionuclide release.

For the fuel itself, the available surface area could range from the geometric surface area of the fuel to the surface area of fuel fragments or to the total surface area assuming all grain boundaries are exposed. Assuming no alteration of the fuel has occurred, the pellets will contain preexisting cracks due to in-reactor operation. The available surface area could be estimated from observations of the degree of cracking generally observed in individual fuel pellets.

Whether grain boundaries are exposed, or will become exposed as a consequence of humid-air oxidation, is much more difficult to decide. Warm, humid-air oxidation leads

to grain boundary oxidation and a loosening of the pellet structure. Whether extensive humid-air oxidation would lead to pellet breakup is not known. However, because the fuel is confined by the residual cladding structure,[1] it is unlikely to crumble to powder. Consequently, a period of humid-air oxidation is not expected to lead to total breakup of the fuel or the multiplication of available surface area by the factor of ~150 anticipated for dry fuel oxidation at high temperatures (see above).

Once temperatures are < 95°C, all of the fuel surface within a failed clad can be assumed to be wetted. This surface area would be the geometric surface area of the pellets multiplied by a factor to account for the degree of cracking of the fuel. Whether this surface area should be multiplied by a second factor to account for penetration down grain boundaries is debatable.

Tests on immersed CANDU fuel conducted over a 19-year period show no significant grain boundary attack of the fuel (Shoesmith et al., 1996). In drip tests, Wronkiewicz et al. (1996) observed grain boundary attack, a process exacerbated when spent fuel, as opposed to UO_2 specimens, was used (Finn et al., 1997). Electrochemical tests have shown that grain boundary attack can occur if the fuel dissolution process is locally fast enough to generate acidity at the dissolution site (due to the hydrolysis of dissolved uranyl ion species, $UO_2^{2+} + H_2O \rightarrow UO_2OH^+ + H^+$, etc.) (Sunder et al., 1996). The different observations of Shoesmith et al. and Wronkiewicz et al. could be attributed to the nature of the test. In the tests by Shoesmith et al., the fuel was immersed in a large volume of water; hence dilution would prevent development of local acidity. In the tests of Wronkiewicz et al., the volume of available water was limited, and any local acidity would be retained at the surface of the fuel. The development of local acidity would be compounded in drip tests on spent fuel by the accompanying radiolysis of humid air. This second influence appears to be the major one, since grain boundary attack on unirradiated UO_2 was not as significant as for spent fuel. Gray et al. (1993, 1994) estimated that the fuel corrosion process penetrates down grain boundaries to a depth

[1] Cladding failure will involve the formation of cracks and/or pits and small crevices; the main body of the

somewhere between 4 to 10 grains, causing about a factor of 15 increase in available surface area. I recommend, therefore, that the calculated fragment surface area be multiplied by a grain boundary factor of 15 to yield the surface area of fuel available for corrosion.

While this may represent the surface area available when the fuel corrosion process begins, the accumulation of deposited corrosion products (in the form of an alteration layer) is likely to reduce the available surface area as corrosion (alteration) progresses. In the absence of drips leading to advective flow, the accumulation of this deposit undoubtedly will block fuel corrosion and radionuclide release, but to what degree has not been determined. In the presence of advective flow across the surface of the fuel (as a consequence of seepage drips), the protectiveness of such deposits likely will be significantly decreased.

THE INTRINSIC DISSOLUTION OF FUEL

Besides the surface area of exposed fuel, many other parameters potentially could influence the intrinsic dissolution (corrosion) rate of fuel. These include: the degree of surface oxidation due to exposure to humid-air conditions, the burnup characteristics of the fuel, the O_2 concentration, water radiolysis, the composition of the groundwater, pH, and temperature. The influence of each of these parameters, to the extent it is known, is discussed briefly below.

The intrinsic dissolution rate is that measured from flow-through tests, since interlaboratory comparisons show these values to be reasonably reproducible (Gray et al, 1994), and comparison to electrochemically predicted values indicates they are free of the influence of the accumulation of alteration products on the corroding fuel surface (Shoesmith et al., 1997b). This rate is appropriate for fuel not exposed to seepage drips.

cladding structure is expected to remain sufficiently intact to constrain the fuel.

Intrinsic dissolution rates were determined for unirradiated UO_2 specimens, although similar experiments have been conducted on spent fuel. Generally, the rates for both are similar, although significant uncertainties remain due primarily to uncertainties in surface areas. Although differences have been observed as a function of fuel burnup, no clear, explicable dependence is observed. The consequences of increased burnup will be: (1) increased radiation fields, (2) increased fission gas release to the fuel/cladding gap, (3) increased grain boundary inventories of radionuclides, and (4) increased surface area due to cracking and porosity. Of these parameters, radiation effects will be dealt with below; (2) and (3) are taken care of in the instant release fraction (discussed above); and (4) has been incorporated in the surface area parameter (discussed above). No evidence exists to show there is any change in the intrinsic corrosion characteristics of UO_2 after in-reactor irradiation; consequently, I do not think it appropriate to include a burnup dependence of the corrosion rate. Indeed, since short-term measurements on spent fuel will include the influence of radiolysis, an influence that will be absent when the fuel is eventually exposed to water in a repository, the inclusion of a burn-up dependence will be erroneous.

Except for its possible influence on exposed fuel surface area (discussed above), a period of humid-air corrosion also is expected to have minor consequences. For the temperatures likely to prevail in a failed waste package (<100°C), oxidation of UO_2/spent fuel will not proceed beyond $UO_{2.33}/UO_{2.4}$, and what evidence exists suggests this degree of oxidation does not significantly change the intrinsic corrosion rate of the fuel. Consequently, the only significant effect of such an oxidation process is likely to be an increase in available surface area, as discussed above.

The dependence of fuel corrosion rate on O_2 concentration has been extensively investigated (Figure DS-2) (Shoesmith et al., 1997b). Electrochemical experiments predict that, with respect to oxygen dissolved in water, the reaction order should be 0.7 under oxidizing conditions. This value is in excellent agreement with the values measured in both single-pass flow-through experiments and other less rigorously controlled experiments (Figure DS-2). A more detailed discussion of these measurements is given elsewhere (Shoesmith et al., 1997b). It is worth noting that the lower apparent

dependence of rate on O_2 concentration for spent fuel can be attributed to radiolytically supported corrosion, which is particularly apparent at low O_2 concentrations. Note also that, in aerated solution, the measured dissolution rate of spent fuel and UO_2 are effectively identical.

The groundwater constituent most likely to accelerate the corrosion of UO_2 is carbonate/ bicarbonate. The reaction order with respect to carbonate has been measured both electrochemically and in single-pass flow-through experiments (Figure DS-3) (Shoesmith et al., 1997b). Except for one set of early electrochemical measurements, when experimental conditions were not well controlled, the reaction order was consistently found to be in the range of 0.4 to 0.6. This range agrees reasonably well with the values predicted from more fundamental electrochemical experiments (0.33 to 0.4).

The influence of carbonate/bicarbonate on fuel corrosion has been investigated in some detail (Shoesmith et al., 1996). The influence can be divided into four categories for conditions when transport limitations are absent.

(1) For concentrations $<10^{-4}$ mol/L^{-1}, carbonate has little influence, and corrosion products accumulate on the fuel surface ($UO_2 \rightarrow UO_2^{2+} \rightarrow UO_3{\cdot}2H_2O$).

(2) For concentrations in the range of 10^{-4} to 10^{-3} mol/L^{-1}, carbonate complexes the uranyl ion and prevents the accumulation of deposits ($UO_2 \rightarrow UO_2^{2+} \rightarrow UO_2(CO_3)_2^{2-}$). In the absence of significant transport of $UO_2(CO_3)_2^{2-}$ away from the fuel surface, as could be the case in a repository, corrosion product deposits eventually will form.

(3) For concentrations in the range of 10^{-3} to 10^{-1} mol/L^{-1} (as in Figure DS-3), carbonate participates kinetically in the corrosion process by forming a surface complex with UO_2^{2+} and accelerating its transfer to solutions ($UO_2 \rightarrow (UO_2CO_3)_{ads} \rightarrow (CO_3)_2^{2-}$). Again, secondary deposits could form in restricted transport conditions.

(4) For concentrations $>10^{-1}$ mol/L^{-1}, a uranyl carbonate deposit forms ($UO_2 \rightarrow (UO_2CO_3)_{solid}$).

According to the results presented by D. Sassani (WFEE Workshop 2), the carbonate concentration in the groundwater after a few thousand years (when fuel exposure would initiate) will be in the range of 10^{-3} to 10^{-4} mol/L^{-1}. Thus, the behavior outlined in (2) and/or (3) above would be expected, and a reaction order with respect to carbonate of ~0.4 would be appropriate (see Figure DS-3).

For the neutral pH range expected to prevail within a failed waste package (7 to 10; D. Sassani, WFEE Workshop 2), the solubility of uranium under oxidizing conditions, and hence the corrosion rate, should vary only slightly. Consequently, only a minor influence of pH is expected.

The influence of temperature on the fuel corrosion process, which is quite uncertain, has been reviewed recently (Sunder and Shoesmith, 1991). The published activation energy values range from 29 to 67 kJ/mol^{-1} depending on the conditions under which it was measured. Three sets of conditions have been investigated.

(1) Acidic conditions, when E_A^* ranges from 50 to 67 kJ/mol^{-1}.

(2) Carbonate solutions, where E_A^* is in the range of 42 to 63 kJ/mol^{-1}.

(3) Non-complexing neutral solutions, when E_A^* is in the range of 29 to 34 kJ/mol^{-1}.

The values for (1) and (2) are in the range expected for chemical dissolution processes, but the values obtained in non-complexing solutions reflect the influence of the accumulation of corrosion products. The thickening of such deposits would prevent the expected increase in corrosion rate with temperature. At the low carbonate concentrations anticipated under repository conditions (10^{-3} to 10^{-4} mol/L^{-1}, D. Sassani,

WFEE Workshop 2), the low range of values obtained for non-complexing neutral conditions probably is appropriate, although it may be judicious to use the higher range obtained for carbonate solutions.

A major fear is that under repository conditions the radiolysis of water to produce any reactive oxidants (e.g., H_2O_2, OH°, $O_2^{\cdot}$) will greatly increase fuel corrosion rates. Figure DS-4 (from Shoesmith et al., 1997b) shows a series of dissolution (corrosion) rates measured as a function of gamma radiation dose rates. The influence of beta radiolysis is expected to be similar. This figure shows that there is a significant influence of gamma radiolysis on fuel corrosion. The horizontal lines, (1) and (2), show the average corrosion rates for zero dose rate in non-complexing aerated solutions (1) and carbonate-containing aerated solutions (2). The horizontal line, marked β/γ, indicates the dose rates expected for spent fuel for the period of 10 to 10^3 years. It is clear that any significant influence of β/γ radiolysis on fuel corrosion will have disappeared in $< 10^3$ years. If containment by the waste package is maintained for a few hundred years, the influence of β/γ radiolysis will be negligible.

A similar insignificance for alpha radiolysis effects could be claimed on the basis of the rates shown in Figure DS-5 (from Sunder et al., 1996b). The rates (from electrochemical predictions) were measured at alpha source strengths well above those anticipated in CANDU fuel (indicated for the period 10 to 10^3 years by the horizontal bar in Figure DS-5). Although α-source strengths for PWR fuels will be greater than those for CANDU fuel, they will not exceed the values used in the measurements. According to the results in Figure DS-5, even for very high α-source strengths the fuel corrosion rate will not exceed that observed in aerated solutions in the absence of any alpha source. It can be concluded, therefore, that the influence of alpha radiolysis will be negligible under repository conditions.

One caveat should be added to this last conclusion. The rates plotted in Figure DS-5 were determined under conditions where the diffusive loss of radiolytic oxidants (O_2 and H_2O_2 in the case of alpha radiolysis) was possible. The influence of alpha radiolysis

could be much greater if the oxidants were confined at the unreacted fuel surface by deposits of corrosion products. At present, there are no data to determine whether such an effect is significant.

The fuel corrosion rate will be influenced significantly by groundwater constituents other than carbonate. Both Wilson and Gray (1990) and Tait and Luht (1997) have shown that the presence of Ca^{2+} and silicates can reduce the corrosion rate substantially. The relevant results of Wilson and Gray (1990) indicate that the rate decreases by as much as two orders of magnitude when Ca^{2+} and silicates are introduced. A similar suppression was observed by Tait and Luht. It is possible that surface phases such as uranophane ($Ca(UO_2)_2\ SiO_3OH)_2 \cdot 5H_2O$) are formed, as observed in the drip tests of Finn et al. (1997). Of the two species, the Ca^{2+} exerts the predominant influence. According to D. Sassani's calculations (WFEE Workshop 2), the water flowing into the repository will be high in Ca^{2+} (as much as $\sim 10^{-2}$ mol/L^{-1}); hence, the corrosion rate of fuel exposed to seepage drips should experience such a suppression (by a factor of 0.01) in corrosion rate.

MODELS TO REPRESENT FUEL DISSOLUTION

Several dissolution models can represent fuel corrosion rates under repository conditions. The first is the model presented by W. Halsey (WFEE Workshop 2):

$$\log R_D = a_o + a.\ T^{-1} + a_2 \log [CO_3] + a_3 \log [O_2] + a_4\ pH + a_5 \log [BU] + \sum a_i x_i x_j \qquad (1)$$

where R_D is the dissolution (corrosion) rate, [BU] represents fuel burnup, a_0 to a_5 are constants determined by a parametric fit to a set of data recorded in single-pass flow-through experiments, and the summation then includes the terms describing interactions between the various parameters in equation (1), also determined from the parametric fit.

An alternative model would have the form of a chemical rate equation based on the dependencies measured for the parameters shown to have a significant influence on fuel corrosion, (surface area, temperature, and oxygen and carbonate concentrations):

$$R_D = k f(A)\, f(T)\, f(GW)\, [CO_3]^p\, [O_2]^q \tag{2}$$

where k is an experimentally determined rate constant, p is in the range of 0.4 to 0.6, and q is in the range of 0.65 to 0.75. The temperature function would have the form of an Arrhenius dependence, and the surface area function would be determined from a cladding failure function modified by the surface area factors discussed above. The factor $f(GW)$ represents the modification of the corrosion rate by groundwater species such as Ca^{2+} and silicates. For fuel not subject to seepage drips, this factor would be 1; for fuel subject to drips it would 0.01.

Because repository carbonate and O_2 concentrations are likely to be invariant once exposure of the fuel occurs, the dependencies on these two terms could be incorporated into the rate constant k (second model) or into the constant a_0 (first model). Also, because I see no need for a pH dependence, and I believe the apparent dependence on burnup (BU) to be spurious, these terms can also be collapsed into a_0 in the first model. Based on these conclusions, the rate equation (using model 2) could be reduced to

$$R_D = k^1\, f(A)\, f(T)\, f(GW) \tag{3}$$

i.e., the intrinsic dissolution rate k^1 would be modified with time only by surface area and temperature ($f(GW)$ is not expected to be time-dependent).

In summary, I conclude that the parameters controlling fuel corrosion are reasonably well understood. The most significant parameters are available water, cladding failures, temperature, and the accumulation of corrosion product deposits (discussed below). The fuel corrosion rate is best represented by a chemical rate equation specified in either equation (2) or (3).

THE INFLUENCE OF CORROSION PRODUCT DEPOSITS (SECONDARY PHASES)

Evidence that fuel corrosion is blocked by the accumulation of corrosion product deposits is extensive and includes immersion tests, flow-through experiments, electrochemical experiments, and natural analogs. Unfortunately, it is presently impossible to quantify this effect, except for the suppression factor of 0.01 (discussed above). Except for the use of the factor f(GW), it is judicious to ignore this effect in present modeling efforts.

A second effect of the formation of such deposits is the immobilization of those radionuclides shown in recent unsaturated spent-fuel tests (Buck et al., 1997) to be incorporated into secondary phases (e.g., ^{237}Np, ^{231}Pa, ^{239}Pu).

Based on the observations in these experiments, the source item for Np (as an example) would be reduced by between four and seven orders of magnitude compared to that (based on Np solubilities) used in TSPA-95. Given the uncertainties about whether colloid transport could occur, particularly in the case of Pu, I recommend that in the short term (for TSPA-VA) the source term based on solubilities should be retained, and that these observations be used only qualitatively to demonstrate the conservatism of the approach used. If, in the longer term (i.e., for TSPA-LA), further evidence in support of these observations can be obtained, then I would recommend their use.

The major uncertainties are: the specification of water chemistry; the mode of water ingress; the performance of the cladding as a barrier; the influence of secondary phases; the importance of alpha radiolysis in causing fuel dissolution; and transport of radionuclides as colloidal species.

1. **Water chemistry.** The chemistry of water entering the failed waste package and how it is influenced by contact with waste package corrosion products and wetting/evaporation cycles within the package should be investigated. To determine

the dissolution behavior of the fuel, it is necessary to know the carbonate, calcium, and silica content of the water. However, of more importance, it is necessary to know the chemistry of the water contacting the cladding. In particular, salinity, pH, and the concentration of potentially aggressive oxidants such as Fe^{III}, Cr^{VI} need to be known if a credible scenario for cladding corrosion is to be specified.

2. **The mode of water ingress.** The mode of water ingress will be a major factor in determining both a cladding failure profile and the degree of mobilization of radionuclides released by fuel dissolution. From the perspective of the cladding, intermittent dripping could lead to the establishment of aggressive occluded chemistries whereas a copious supply would prevent it. For the mobilization of radionuclides, the more frequent and voluminous the water supply, the faster transport will be.

3. **Cladding failure function.** The lack of information on the corrosion of Zircaloy cladding makes the specification of a credible failure function very difficult. A substantial experimental program is required if exclusive credit for cladding is to be taken.

4. **Secondary phases.** A much more detailed understanding of the function of secondary phases is required, since very significant retarding influences on fuel dissolution and radionuclide release could then be incorporated into PA calculations. The influence of the accumulation of secondary phase deposits on the fuel dissolution rate should be determined since this will affect the release rate of all radionuclides within the fuel matrix. Of even greater importance is the ability of these phases to incorporate and immobilize radionuclides such as ^{237}Np and ^{239}Pu. Good grounds exist to suspect that the TSPA-95 treatment of Np (in particular) is extremely conservative. The advantages to be gained by characterizing secondary phases and determining their ability to incorporate radionuclides would be large.

5. **Colloids.** The fate of radionuclides such as ^{239}Pu depends on the efficiency of their immobilization once they have been released from the fuel. This would be expected to be high and the only significant transport pathway would be via colloidal transport. A much more detailed understanding of the possibilities for colloidal transport is required.

6. **Alpha radiolysis effects.** While the influence of alpha radiolysis on fuel corrosion is expected to be minor, this remains to be conclusively demonstrated. In particular, the possibility of the production of radiolytic oxidants at the fuel/deposit interface should be investigated.

RECOMMENDED ACTIVITIES TO REDUCE UNCERTAINTIES

1. Composition of the water contacting the cladding and the waste form needs to be established.
2. Mode of water ingress required in order to determine the cladding corrosion function and the ability of deposits to block fuel corrosion.
3. A cladding failure function that incorporates the possibility of localized corrosion needs to be developed.
4. The possibility that deposited secondary phases could block fuel dissolution and retain radionuclides needs to be investigated.
5. The ability of colloids to retain and transport radionuclides should be investigated.
6. The importance of alpha radiolysis for fuel dissolution remains uncertain.

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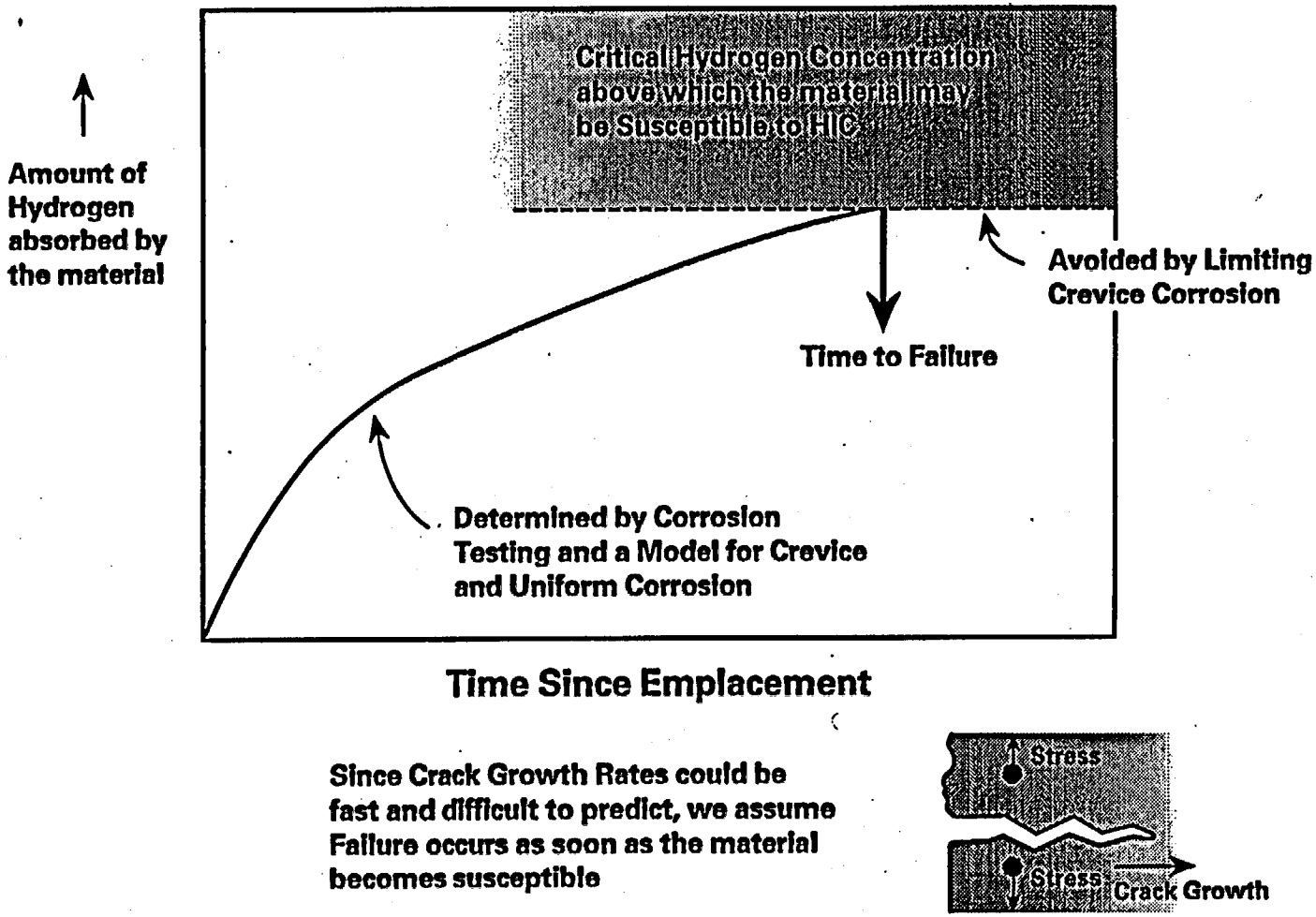


Figure DS-1 The path to failure by hydrogen absorption due to localized corrosion and the criterion to decide when failure should be taken to have occurred.

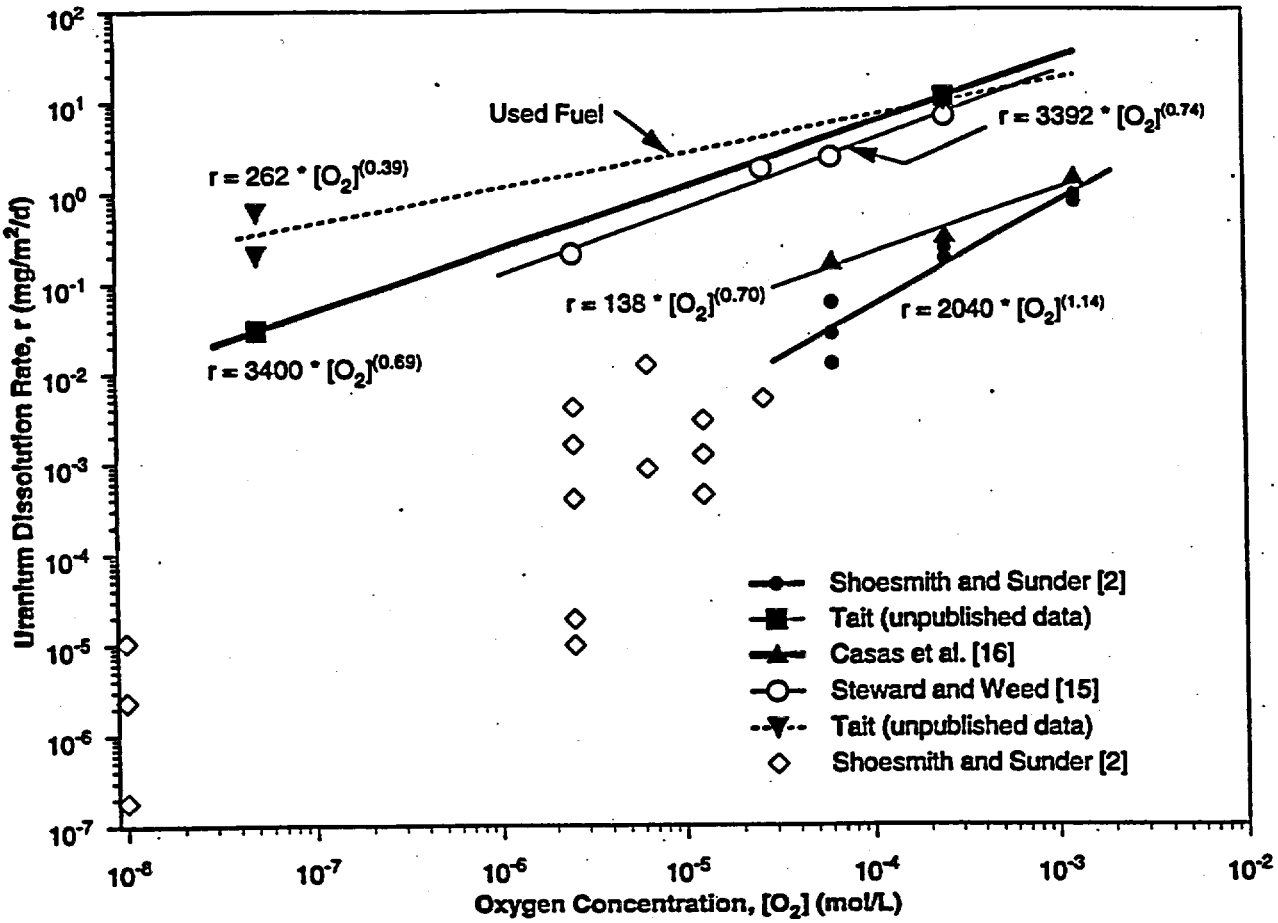


Figure DS-2 Dissolution (corrosion) rates for UO_2 and spent fuel as a function of oxygen concentration at 25°C (from Shoesmith et al., 1997b). The equations represent the best fit to the data. Shoesmith and Sunder (1991); predicted using an electrochemical model for oxygenated 0.1 $mol.L^{-1}$ $NaClO_4$ (pH = 9.5); Tait (unpublished results); measured using the single pass flow through technique in 0.01 $mol.L^{-1}$ /0.1$mol.L^{-1}$ $NaClO_4$ (pH =9.5); Casas et al.(1993); measured in oxygenated $MgCl_2$ brine (pH = 4.5); Steward and Weed (1994); measured using the single pass flow through technique in 0.02 $mol.L^{-1}$ $NaHCO_3$ (pH = 9).

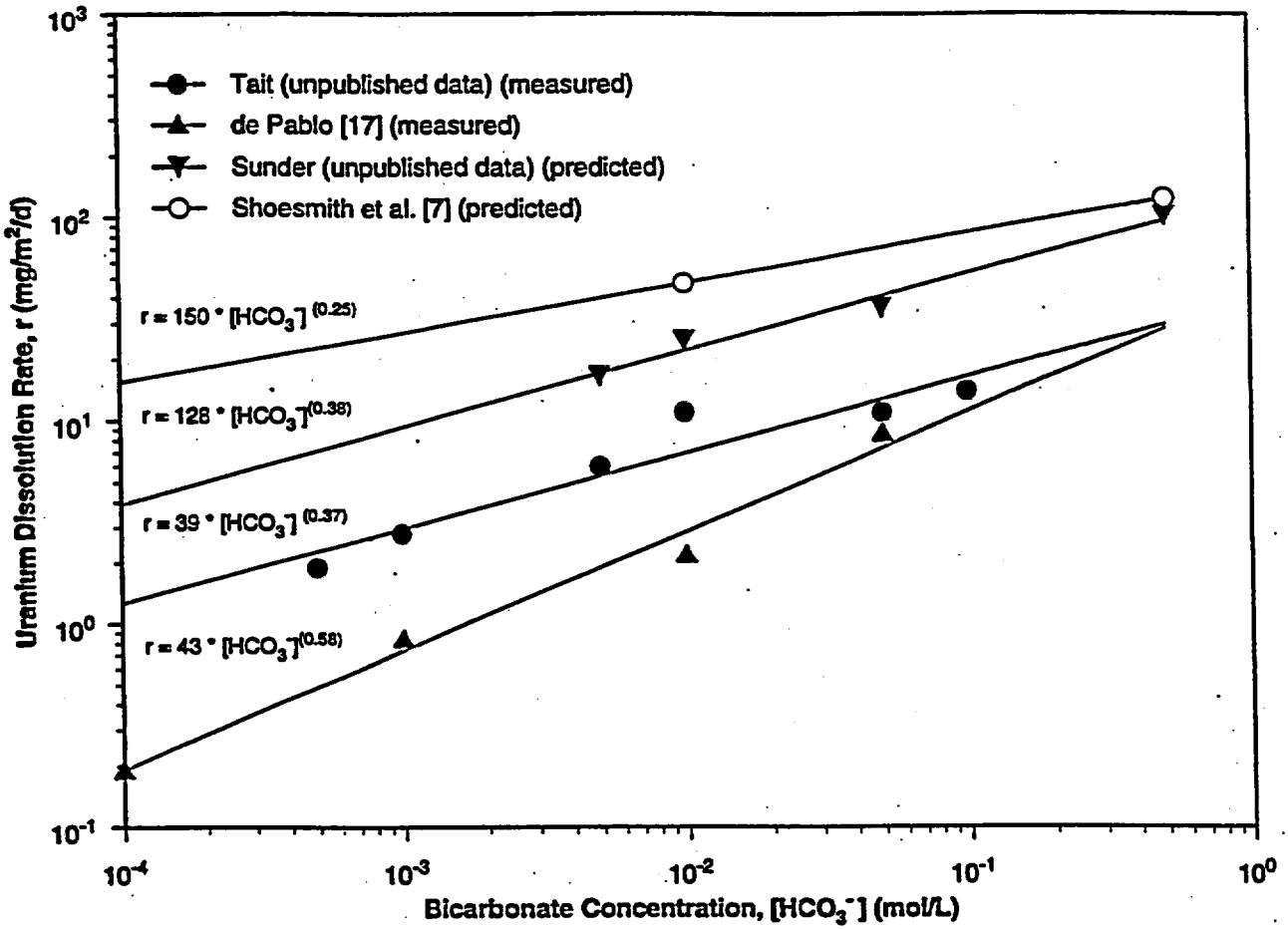


Figure DS-3 Dissolution (corrosion) rates for UO_2 as a function of bicarbonate ion concentration in aerated solution at 25°C (from Shoesmith et al., 1997b). The equations show the best fits. Tait (unpublished results); measured using the single pass flow through technique in 0.1 $mol.L^{-1}$ NaCl (pH = 9); De Pablo et al. (1997); measured in 0.1 $mol.L^{-1}$ NaCl (pH = 9); Sunder and Shoesmith (unpublished results) and Shoesmith and Sunder (1989); predicted using an electrochemical model for aerated 0.1 $mol.L^{-1}$ $NaClO_4$ (pH = 9.5).

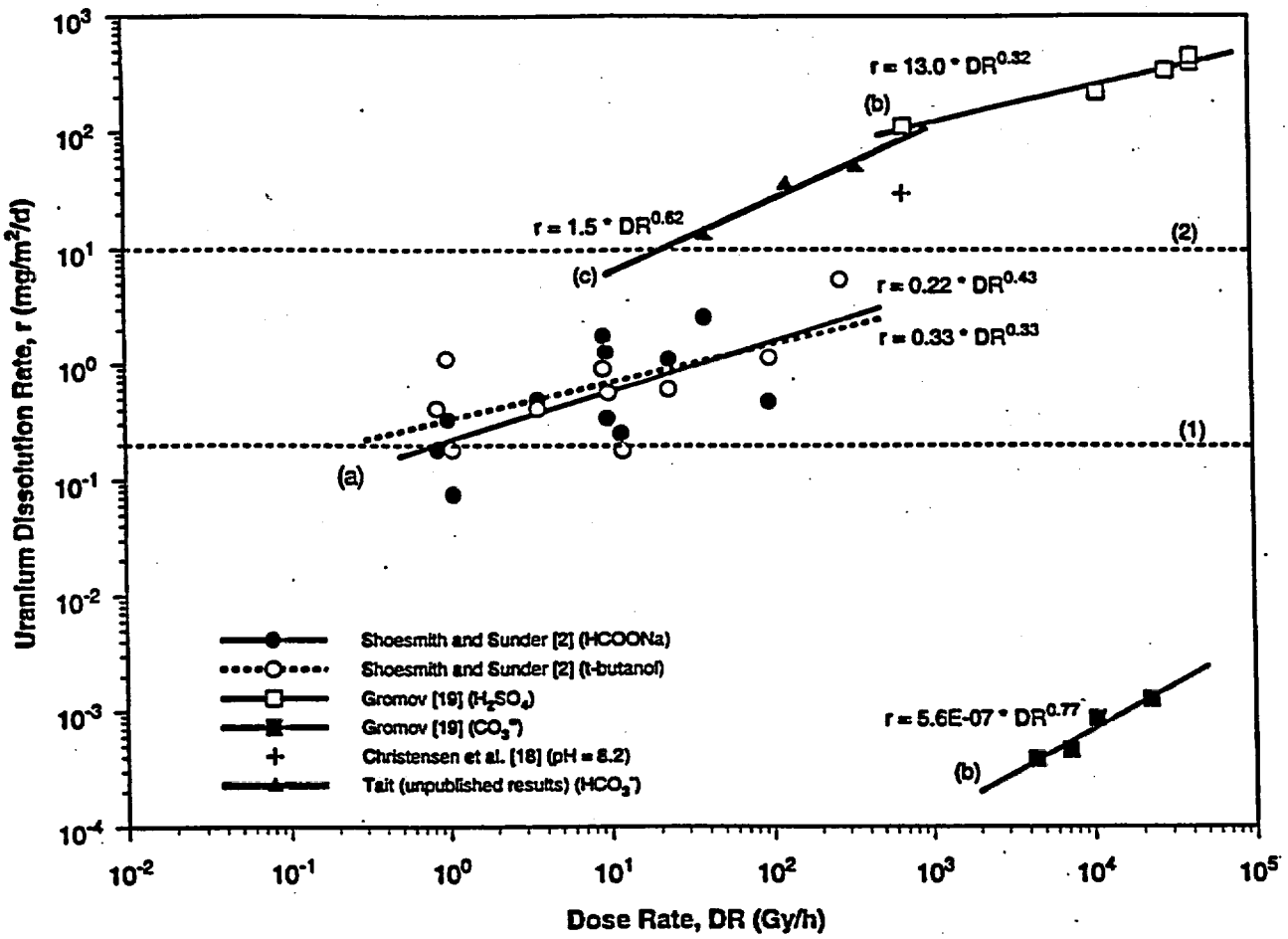


Figure DS-4 Dissolution (corrosion) rates for UO_2 in irradiated, aerated/oxygenated solutions as a function of the gamma dose rate. The horizontal dashed lines show the rate predicted for unirradiated aerated solution (1) (from Figure DS-2) and the rate measured in flow through experiments in bicarbonate solution (2) (from Figure DS-3). The solid horizontal line shows the approximate range of beta/gamma dose rates expected over the repository period 10 (high doses) to 10^3years (low doses). Shoesmith and Sunder (1991); Predicted using an electrochemical model for aerated 0.1 mol.L^{-1} $NaClO_4$ (pH = 9.5) containing added HCOONa (solid points) or t-butanol (open points); Gromov (1981); measured in acidic sulphate solution (pH = 1) (open squares) or alkaline carbonate solution (pH = 10) (closed squares); Tait (unpublished results); measured using the single pass flow through technique in 0.1 mol.L^{-1} NaCl and 0.01 $NaHCO_3$ (pH = 8.5); Christensen et al. (1990); measured in oxygenated solution (pH = 8.2) (single crossed data point).

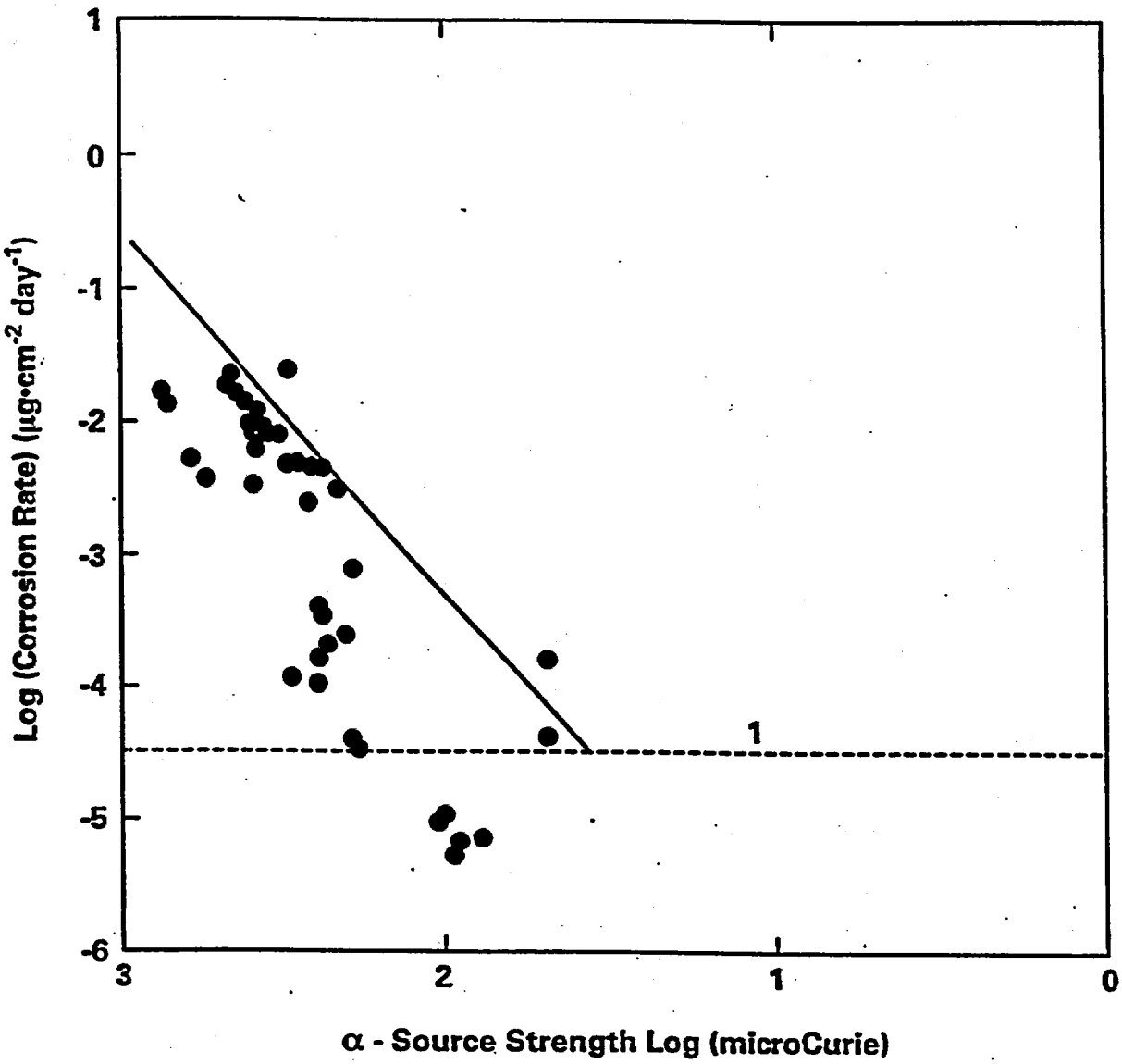


Figure DS-5 Dissolution (corrosion) rates of UO_2 as a function of alpha source strength measured in a 0.1 mol.L^{-1} $NaClO_4$ (pH = 9.5) solution (from Sunder et al., 1997). The horizontal dashed line corresponds to the threshold above which kinetically controlled oxidative dissolution (corrosion) of UO_2 occurs. The solid line shows the approximate range of alpha source strengths expected over the range 10 (high strength) to 10^3 years (low strength). The shaded range shows the range of rates expected in aerated solutions in the absence of an alpha source.

APPENDIX E
QUALITY ASSURANCE

QUALITY ASSURANCE

The Waste Form Degradation and Radionuclide Mobilization Expert Elicitation Project was completed under the Quality Assurance Program for the Civilian Radioactive Waste Management and Operating Contractor (CRWMS M&O). The process and procedures for M&O staff conducting the activity were defined in the Project Plan for this expert elicitation project. Section 2.0 of this report summarizes the process for eliciting expert judgments. As discussed in Section 2.0, formal guidance for the process of expert elicitation has been established and this guidance has been successfully applied in other comparable assessments.

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Director's Program Review

PRESENTED TO

Lake Barrett, Acting Director

OCRWM

May 7, 1998

AGENDA

Director's Program Review
Thursday, May 7, 1998
Videoconference Rooms: M&O Contractor (Dunn Loring),
DOE/Forrestal, Room GF-277, and YMSCO Blue Room

Time (PST)	Subject	Presenter
7:00 AM - 7:05 AM	Recognition of Visitors	Conner
7:05 AM - 7:10 AM	Opening Remarks	Barrett
7:10 AM - 7:20 AM	Program Status Overview Program Performance Status	Rousso
7:20 AM - 8:20 AM	YMSCO Overview Viability Assessment: TSPA Base Case YMP Performance Measurement	 Jones VanLuik Kozai
8:20 AM - 8:40 AM	WAST Project Overview WAST Performance Measurement	 Shelor Bokhari
8:40 AM - 9:00 AM	Quality Assurance Overview QA Performance Measurement	Horton
9:00 AM - 9:20 AM	Program Management and Administration Overview PM&A Performance Measurement	 Rousso Trebules
9:20 AM - 9:30 AM	Review of the Day's Action Items	Conner
9:30 AM - 9:40 AM	Questions from Visitors	All
9:40 AM - 9:55 AM	Lunch at Seats	
9:55 AM - TBD	Executive Session	

OVERVIEW

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Director's Program Review

Program Overview

PRESENTED TO

LAKE BARRETT, Acting Director
OCRWM

PRESENTED BY

SAM ROUSSO, Director
Office of Program Management
and Administration

May 7, 1998

5/6/98
DCB5073

CHANGES SINCE LAST DPR

- Approved Level 1 Changes
  - Initial Issuance of OCRWM Strategic System Management Policy
- Pending Changes
  - Issuance of CRWMS System Requirements Document (CRD), Rev. 4

ACCOMPLISHMENTS
March-April

- Viability Assessment
- Site Recommendation/License Application
- CISF TSAR
- Strategic System Management Policy

LEVEL 1
MILESTONE STATUS

WBS	ID Number	Description	Baseline Date	Forecast/ Actual Date
3.1.4	WT314M8003	Issue NWPA Sec. 180 (c) Notice of Revised Proposed Policy and Procedures	02/01/98	04/30/98(A)
1.2.5	M1AP	Publish 10CFR960 Final Rule	08/31/98	TBD*
1.2.5	M1AT	OCRWM Accepts Viability Assessment	09/04/98	09/04/98
3.1.1.1	WT311M7004	Issue Final RFP for Acquisition of RSA WA&T Services	TBD	08/01/01*

* Date change since last DPR

U.S Department of Energy
Cost Performance Report - Work Breakdown Structure (Format 1)

1. Title	2. Reporting Period	3. Dollars Expressed In Thousands Escalated
Waste Management System	01-Mar-1998 to 31-Mar-1998	6. Report Date
4. Participant Name & Address	5. Cost Plan Date	7. Manager's Signature
OCRWM		

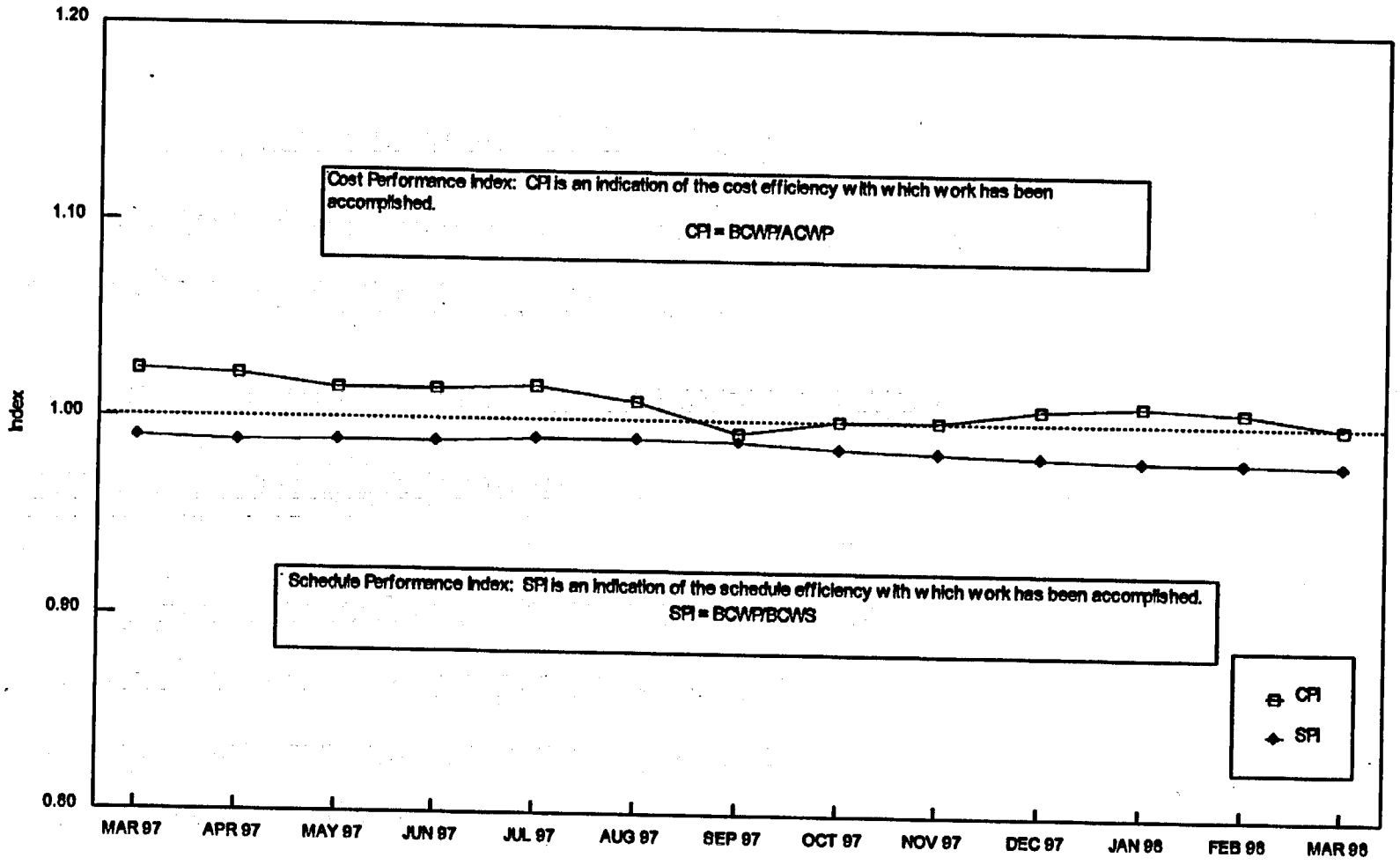
8. WBS Elements	Current Period					FY96-98 to Date					FY96-98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
WASTE MANAGEMENT SYSTEM													
1.O Mined Geologic Disposal System	26322	24608	30282	-1714	-5674	641250	630767	630112	-10483	655	787550	802570	-15020
3.O WAST Project	500	558	524	58	34	44964	45050	42207	86	2843	47917	45616	2301
9.O Program Management & Administration	2138	2135	1976	-3	159	86398	86395	80032	-3	6363	99135	94046	5089
9. Contractor Budget Base	28960	27301	32782	-1659	-5481	772612	762212	752351	-10400	9861	934602	942232	-7630
10. Financial & Tech. Assissatance	2278	2278	2278	0	0	69179	69179	86890	0	-17711	89132	106843	-17711
11. Undistributed Budget											12865	12865	0
13. Fee											34787	34787	0
14. Baseline Adjustment/Contingency											34491	34491	0
15. Perf. Measurement Baseline	31238	29579	35060	-1659	-5481	841791	831391	839241	-10400	-7850	1105877	1131218	-25341
16. Management Reserve											8049	8049	0
17. Other											13818	13818	0
18. Program Budget Base	31238	29579	35060	-1659	-5481	841791	831391	839241	-10400	-7850	1127744	1153085	-25341

U.S Department of Energy
Cost Performance Report - Work Breakdown Structure (Format 1)

1. Title	2. Reporting Period	3. Dollars Expressed in Thousands Escalated
Waste Management System	01-Mar-1998 to 31-Mar-1998	6. Report Date
4. Participant Name & Address	5. Cost Plan Date	7. Manager's Signature
OCRWM		

8. WBS Elements	Current Period					FY98 to Date					FY98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
WASTE MANAGEMENT SYSTEM													
1.O Mined Geologic Disposal System	26322	24608	30282	-1714	-5674	152160	145136	140835	-7024	4301	298460	313293	-14833
3.O WAST Project	500	558	475	58	83	2507	2593	2292	86	301	5460	5409	51
9.O Program Management & Administration	2138	2135	1976	-3	159	12242	12217	10943	-25	1274	24979	24773	206
9. Contractor Budget Base	28960	27301	32733	-1659	-5432	166909	159946	154070	-6963	5876	328899	343475	-14576
10. Financial & Tech. Assissatance	2278	2278	2278	0	0	19352	19352	19352	0	0	39305	39305	0
11. Undistributed Budget											250	250	0
13. Fee											12795	12742	53
14. Baseline Adjustment/Contingency											3529	0	3529
15. Perf. Measurement Baseline	31238	29579	35011	-1659	-5432	186261	179298	173422	-6963	5876	384778	395772	-10994
16. Management Reserve											4273	3850	423
17. Other											9630	9630	0
18. Program Budget Base	31238	29579	35011	-1659	-5432	186261	179298	173422	-6963	5876	398681	409252	-10571

FY 1997/1998 PROGRAM CPI/SPI GRAPH



FY 97/98	MAR 97	APR 97	MAY 97	JUN 97	JUL 97	AUG 97	SEP 97	OCT 97	NOV 97	DEC 97	JAN 98	FEB 98	MAR 98
CPI	1.024	1.022	1.016	1.016	1.017	1.010	0.994	1.001	1.001	1.007	1.010	1.007	1.000
SPI	0.990	0.988	0.989	0.989	0.991	0.991	0.990	0.987	0.985	0.983	0.982	0.982	0.981

5/6/98
DCB5073

SIX MONTH LOOK AHEAD

- **YMP Milestones**
  - MGR Cost Estimate
  - MGR-VA Design Technical Report
  - Excavation of the ECRB Cross Drift
  - OCRWM Accepts Viability Assessment
- **WAST Milestones**
  - RFP for Acquisition of RSA WA&T Services
- **Program Deliverables**
  - FY 2000 OMB Request to CFO
  - TSLCC and Fee Adequacy Report

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Director's Program Review

Yucca Mountain Project Overview

PRESENTED TO

Lake Barrett, Acting Director

OCRWM

PRESENTED BY

Susan B. Jones

Acting Deputy Project Manager

May 7, 1998

PRIORITY ACTIVITIES

- Viability Assessment
- Site Recommendation/License Application
  - Enhanced Characterization of the Repository Block
  - Busted Butte Unsaturated Zone Test Facility
- EIS, Environment, Safety and Health
- Administration and Asset Management

VIABILITY ASSESSMENT

- **Goals:**
  - Provide the Secretary and Congress with a defensible Viability Assessment in 1998 that meets the requirements of the FY 1997 Appropriations
- **Components of the VA**
  - A preliminary design concept for the critical elements for the repository and waste package
  - A total system performance assessment, based upon a design concept and available scientific data and analyses describing the expected behavior of the proposed repository
  - A plan and cost estimate for the remaining work required to complete a license application
  - An estimate of the costs to construct and operate the repository in accordance with the design concept

VIABILITY ASSESSMENT, Cont.

- **Review Schedule:**

REVIEW	KICKOFF MEETING	COMMENT PERIOD CLOSES	COMMENT RESOLUTION MEETING	NEW MASTER FILES POSTED TO THE STORYBOARD
M&O/YMSCO Review:				
Volume 1, Site Description	April 21	May 1	May 12-13	May 15
Volume 4, LA Plan	May 5	May 15	May 27-28	June 4
Volume 2, Design	May 19	May 29	June 9-10	June 12
Executive Summary	May 19	May 29	June 11-12	June 16
Volume 3, TSPA-VA	June 2	June 12	June 18-19	June 23
Volume 5, Cost	June 2	June 12	June 23-24	June 26
DOE-Wide Review	July 1	July 13	July 21-23	July 28

VIABILITY ASSESSMENT, Cont.

- **Status of Design Issues:**
  - Eleven accepted, five under review, four to be submitted this month

EVENT	DESCRIPTION		BASELINE DATE	SUBMITTED FOR REVIEW	ACCEPTED	STATUS
SE9220M3	Issue #9:	Strategy for Mapping the Repository	30-Jan-98	30-Jan-98	05-Mar-98	Accepted
SE9120M3	Issue #10:	Post Closure Performance Standards	30-Jan-98	30-Jan-98	05-Mar-98	Accepted
SE9130M3	Issue #15:	Additional Waste Forms	30-Jan-98	30-Jan-98	05-Mar-98	Accepted
WP9320M3	Issue #17:	Waste Package Materials	30-Jan-98	30-Jan-98	09-Mar-98	Accepted
RP9420M3	Issue #6:	Retrievability Concept	30-Mar-98	30-Mar-98	27-Apr-98	Accepted
RP9430M3	Issue #7:	Conf. High Volume/Long Period Waste	30-Mar-98	30-Mar-98	27-Apr-98	Accepted
RP9440M3	Issue #11:	Viability of Ungd Remote Control Con	30-Mar-98	30-Mar-98		Under Review
SE9240M3	Issue #13:	Repository Seals Req. & Concepts	30-Mar-98	30-Mar-98		Under Review
RP9450M3	Issue #19:	Subsurface Development	30-Mar-98	30-Mar-98	30-Apr-98	Accepted
RP9460M3	Issue #20:	Surface Development	30-Mar-98	30-Mar-98	30-Apr-98	Accepted
RP9470M3	Issue #21:	Site Development	30-Mar-98	30-Mar-98	30-Apr-98	Accepted
SE9110M3	Issue #1:	Thermal Loading	31-Mar-98	30-Mar-98		Under Review
SE9210M3	Issue #2:	EBS Performance Enhancement	30-Apr-98	30-Apr-98		Under Review
WP9310M3	Issue #16:	Waste Package Size & Weight	30-Apr-98	30-Apr-98		Under Review
RP9410M3	Issue #4:	Emplacement Drift Ground Support	08-May-98	26-Feb-98	31-Mar-98	Accepted
SE9230M3	Issue #8:	Disposal of Site Generated Waste	14-May-98	30-Mar-98	27-Apr-98	Accepted
SE9250M3	Issue #14:	RSA/ISF Interface	14-May-98			
WP9330M3	Issue #3:	Criticality Control	29-May-98			
SE9260M3	Issue #5:	Performance Confirmation Concepts	29-May-98			
SE9140M3	Issue #18:	Design Basis Model	29-May-98			

ACCOMPLISHMENTS
March-April

Viability Assessment

- Initiated review of Volume 1, Site Description, on April 21
- Placed 11 VA supporting documents on the YMP Homepage as requested by AML (joint efforts of the M&O and KENROB staffs)
- Completed VA process models for:
  - Container material Alloy C-22 performance
  - Ceramic coating degradation
  - Waste form dissolution and radionuclide release
  - Waste form cladding degradation

ACCOMPLISHMENTS
March-April

Viability Assessment, cont.

- Completed Draft Disposability Interface Specification for DOE-Spent Nuclear Fuel (SNF), Rev. 0
- Completed Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase II Degraded Codisposal Waste Package Internal Criticality, Rev. 01
- Planned and participated in an Appendix 7 Meeting, Richland, WA, on DOE-Owned SNF — April 29

ACCOMPLISHMENTS
March-April

Viability Assessment, cont.

- Accepted the Site Gas/Liquid Systems Technical Report, RP742DM3. Identifies the site-wide gaseous and liquid systems needed to support all site functions including shipping cask receipt and washdown, waste form transfers, disposable container loading and sealing, decontamination, lag storage, low-level waste treatment, etc. — March 5

- Accepted the Site Layout Analysis, RP742BM3. Analyzes and defines the layout of all site buildings to most effectively and strategically locate surface facility functions — March 5

ACCOMPLISHMENTS
March-April

Viability Assessment, cont.

- Accepted the Site Communications and Control Systems Technical Report, RP742AM3. Defines the primary site communications system functions (voice, video, and data) needed for site systems including phone, security, emergency response, command and control, and transportation dispatch — March 23

- Accepted the Site Electrical Systems Technical Report, RP742CM3. Identifies the surface site-wide electrical systems needed to support repository operations including power distribution, lighting, emergency/standby power, and uninterruptable power systems — March 24

ACCOMPLISHMENTS
March-April

Viability Assessment, cont.

- Received for review the Final Viability Assessment MGDS Test and Evaluation Plan, SE420M3 — March 31
- Received for review the Transportation to MGDS Interface Control Document, SE168M3 — March 31

SITE RECOMMENDATION/LICENSE APPLICATION

- **Goals:**
  - Develop management plan, logic network, and schedule for completing the components of the Site Recommendation (SR)
    - Repository description with preliminary engineering specifications
    - Waste form or packaging description; relationship with site
    - Discussion of site characterization data relating to safety of site
    - Final Environmental Impact Statement
    - NRC Preliminary Comments on Sufficiency of Site Characterization

SITE RECOMMENDATION/LICENSE APPLICATION, cont.

- **Goals, cont.**
  - Views of Governor, Legislature, and Tribes with Secretary's Responses
  - Any other information Secretary considers appropriate
  - Nevada statement of impacts likely to result from site characterization
  - Continue site characterization activities that support SR and License Application (LA)
  - Develop logic network and schedule for the LA, in the event that the President approves, and Congress designates, the Yucca Mountain site for development as a repository

ACCOMPLISHMENTS
March-April

Site Recommendation/License Application

- Enhanced Characterization of the Repository Block
  - Completed design of the Cross Drift, SCM040M3 (5 days ahead of schedule) — March 6
  - Completed equipment acquisition and setup for Cross Drift Ventilation Testing and Monitoring, SC820M3 (1 day ahead of schedule) — March 17
  - Completed TBM assembly and placement in the launch chamber; completed installation of support utilities and equipment
    - Initiated TBM excavation of the Cross Drift, M2AK (17 days ahead of schedule) (Level 2) — March 31

ACCOMPLISHMENTS
March-April

Site Recommendation/License Application, cont.

- Busted Butte Unsaturated Zone Test Facility
  - Phase 1a Testing: Completed air permeability, video, and neutron logging of test boreholes. Installed data collection instrumentation and set up data recording station
    - Completed a dry run — March 26
    - Initiated tracer injection — April 2
  - Phase 1b Testing: Completed air permeability, video, neutron and ground penetrating radar logging of test boreholes. Installed data collection instrumentation and set up data recording station
    - Completed a dry run — April 17
    - Initiated tracer injection — April 24

ACCOMPLISHMENTS
March-April

Site Recommendation/License Application, cont.

- Busted Butte Unsaturated Zone Test Facility, cont.
  - Phase 2 Testing: Completed drilling/coring of holes to be used for tracer collection, injection, and electrical resistivity tomography (included additional boreholes). Performed air permeability, video, and neutron logging of test boreholes. Submitted permit to the State of Nevada for tracers to be injected during Phase II Testing

March 19 (2 months ahead of schedule)

EIS & ENVIRONMENT, SAFETY, AND HEALTH

- **Goals:**
  - Provide protection for the environment, and health and safety of project employees, the public, and project facilities
  - Provide ES&H oversight to site characterization activities, and meet applicable DOE and regulatory requirements, including the National Environmental Policy Act

ACCOMPLISHMENTS
March-April

EIS & Environment, Safety, and Health

- EIS Status:
  - Briefed EIS progress to NWTRB staff member — April 21
  - Continued data collection and impact analysis, including coordination of TSPA input, for preparation of preliminary draft EIS chapters. Activities remain on schedule for July 1999 draft EIS release and public comment period

ACCOMPLISHMENTS
March-April

EIS & Environment, Safety, and Health, cont.

- ES&H Accomplishments:
  - Continued preparations for technical, scientific and administrative support for the Water Appropriations Permit Hearing
  - Completed an emergency management exercise on April 28. A simulated ESF emergency resulted in an evacuation and subsequent reactions by joint underground rescue teams, emergency medical and fire protection elements
  - Initiated development of the OCRWM/YMSCO Functions, Responsibilities and Authorities Manual for completion by the July 1998 date required in the Departmental Strategic Plan

ACCOMPLISHMENTS
March-April

EIS & Environment, Safety, and Health, cont.

- ES&H Accomplishments, cont.
  - Conducted a Tribal Update Meeting with representatives from the 17 tribes and organizations with cultural and historical ties to Yucca Mountain. Provided an update on Project activities, the EIS, cultural resources, and the Native American programs — May 1-2
  - Delivered the Underground Injection Control Report, SS9830, to the State of Nevada, Division of Environmental Protection, in accordance with YMP permit requirements — April 27

ADMINISTRATION AND ASSET MANAGEMENT

- **Goals:**
  - Provide information management support to VA. During this reporting period, critical activities have included performing software analyses and providing information management systems, including records management processes which support the traceability, retrievability, and distribution of data supporting VA
  - Provide safe and environmentally sound construction to support SR/LA

ACCOMPLISHMENTS
March-April

Administration and Asset Management

- In collaboration with RW/Information Management, submitted Information Technology (IT) UNICALL budget information, prepared the FY 1999 IT planning guidance to address new requirements imposed by OMB, and responded to DOE/HQ requests related to Year 2000
- Received the Integrated Safety Management System (ISMS) Implementation Plan. The ISMS description submittal is planned for September
- Provided construction management of the Cross Drift and Busted Butte facilities to construct test areas for data accumulation

ACCOMPLISHMENTS
March-April

Administration and Asset Management, cont.

- Issued revised computer security policies and approved WEB access configuration based on National Institute of Standards and Technology guidelines for initial risk assessment
- To date, 51% of 216,000 records planned for reprocessing in FY 1998 have been completed

ACCOMPLISHMENTS
March-April

Administration and Asset Management, cont.

• Completed transfer of the TRW Contract as planned	April 1
• Accepted the Performance Evaluation and Measurement Plan (PEMP) 11-4 Letter Report, BMI98M3 (analyses improve documentation preparation and management processes)	April 28

CHANGES SINCE LAST DPR
February-March

Changes Approved in February - March:

CR 98/007	Modify WBS 1.2.6 and Associated Project Baseline to add New 4th Level WBS Elements for ECRB
CR 98/008	Add and Adjust Project Baseline Work Scope to Address Performance Evaluation and Measurement Plan (PEMP) No. 11, Goal 4
CR 98/009	Level 2 Baseline of the Mined Geologic Disposal System Requirements Document (MGDS-RD), Rev. 3
CR 98/010	In Situ Stress Tests
CR 98/011	Addition of New Workscope and Budget for AECL in WBS 1.2.3.1 for Participation of AECL in an Expert Elicitation and Cl-36 Peer Review Panel
CR 98/012	Provide Support to International High-Level Waste Management Conference
CR 98/013	Add Baseline Work Scope to Complete USW WT-24 Borehole Testing

CHANGES SINCE LAST DPR
February-March

Changes Approved in February - March, Cont.:

CR 98/014	Increase Baseline Work Scope/Budget for East/West Drift Hydrologic Predictive Report to Address Additional Level 3 Milestone Criteria
CR 98/015	Correction of Errors in FY98 Cost and Schedule Baseline

Changes Approved in April:

CR 98/016	Add Baseline Work Scope/Budget for Support to Line Organizations for Ongoing Documentation Issues and Backlog
CR 98/017	Add Baseline Work Scope/Budget to Conduct Ground Motion Calculations for Upper and Lower Bound Velocity Profiles
CR 98/018	Add Baseline Work Scope/Budget to Generate Draft Appendix E of the Environmental Impact Statement
CR 98/019	Add Baseline Work Scope/Budget to Update the State-Level Accident Risk of Interstate Freight Transportation
CR 98/020	Add Baseline Work Scope/Budget to Conduct Terrorism Impacts Analysis

FY 1998 BASELINE
LEVEL 1 MILESTONES

WBS	ID Number	Description	Baseline Date	Forecast/ Actual(A) Date
1.2.5	M1AP	Publish 10CFR960 Final Rule	08/31/98	08/31/98
1.2.5	M1AT	OCRWM Accepts Viability Assessment	09/04/98	09/04/98

FY 1998 BASELINE
LEVEL 2 MILESTONES

WBS	ID Number	Description	Baseline Date	Forecast/ Actual(A) Date
1.2.1	M2FZ	Complete MGDS-VA Design Technical Report	08/21/98	09/01/98
1.2.1	M2KR	Complete MGDS Cost Estimate	08/28/98	08/25/98
1.2.3	M2EP	Initiate Drift Scale Thermal Test	12/08/97	12/03/97 (A)
1.2.3	M2EM	Initiate Test In Situ (Alcove 7)	12/18/97	12/18/97 (A)
1.2.3	M2EV	Complete PSHA	03/25/98	06/01/98
1.2.3	M2FC	Near Field Models Acceptance	04/23/98	06/05/98
1.2.5	M2EG	YMSCO Issues PR 16	10/31/97	10/21/97 (A)
1.2.5	M2ER	Modeling Scenario & Abstract/Peer Review 2nd Report	12/15/97	02/27/98 (A)
1.2.5	M2EX	YMSCO Submits Draft Final 10CFR960 to OCRWM	02/09/98	02/09/98
1.2.5	M2YY	Submit Final Draft 10CFR960 Rule to NRC	03/10/98	03/10/98
1.2.5	M2FA	YMSCO Issues PR 17	04/29/98	06/03/98
1.2.5	M2FM	Complete TSPA Peer Review Interim Report #3	07/15/98	07/31/98

FY 1998 BASELINE
LEVEL 2 MILESTONES

WBS	ID Number	Description	Baseline Date	Forecast/ Actual(A) Date
1.2.5	M2FT	Complete TSPA-VA Analysis	08/21/98	08/21/98
1.2.5	M2FW	Complete LA Plan - VA Tech Component	08/21/98	08/21/98
1.2.5	M2FR	Complete Technical Record for VA	08/28/98	08/28/98
1.2.5	M2KT	YMSCO Accepts Viability Assessment	08/28/98	09/04/98
1.2.6	M2EE	Complete SGDF Excavation	10/31/97	10/08/97 (A)
1.2.6	M2AS	Start Excavation of Busted Butte Facility	12/05/97	12/18/97 (A)
1.2.6	M2AH	Start Launch Chamber Excavation	12/15/97	12/12/97 (A)
1.2.6	M2AJ	Complete Launch Chamber Excavation to Station 00+26	02/05/98	02/04/98 (A)
1.2.6	M2AY	Complete Excavation of the Busted Butte Facility	02/06/98	01/30/98 (A)
1.2.6	M2AK	Start TBM Operation of the Cross Drift	04/17/98	03/31/98 (A)
1.2.6	M2AQ	Complete Excavation Across Repository Block to Station 25+00	08/24/98	08/06/98
1.2.9	M2FE	YMP Plan Annual Update	09/30/98	09/30/98

DPR ACTION ITEMS

Action Item #:	2
Action Item:	YMSCO should finalize a plan for getting the completed VA documents on the OCRWM Home page as they are completed
OPR:	YMSCO
Status:	Closed

DPR ACTION ITEMS

Action Item #:	3
Action Item:	Provide RW-1 a detailed schedule covering the period from now to the completion of the report for the Repository Modular Design, Constrained Funding, Early Receipt Report
OPR:	YMSCO
Status:	Change request scheduled for submittal on May 8

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Viability Assessment: TSPA Base Case

PRESENTED TO

Lake Barrett, Acting Director

OCRWM

PRESENTED BY

Abe VanLuik

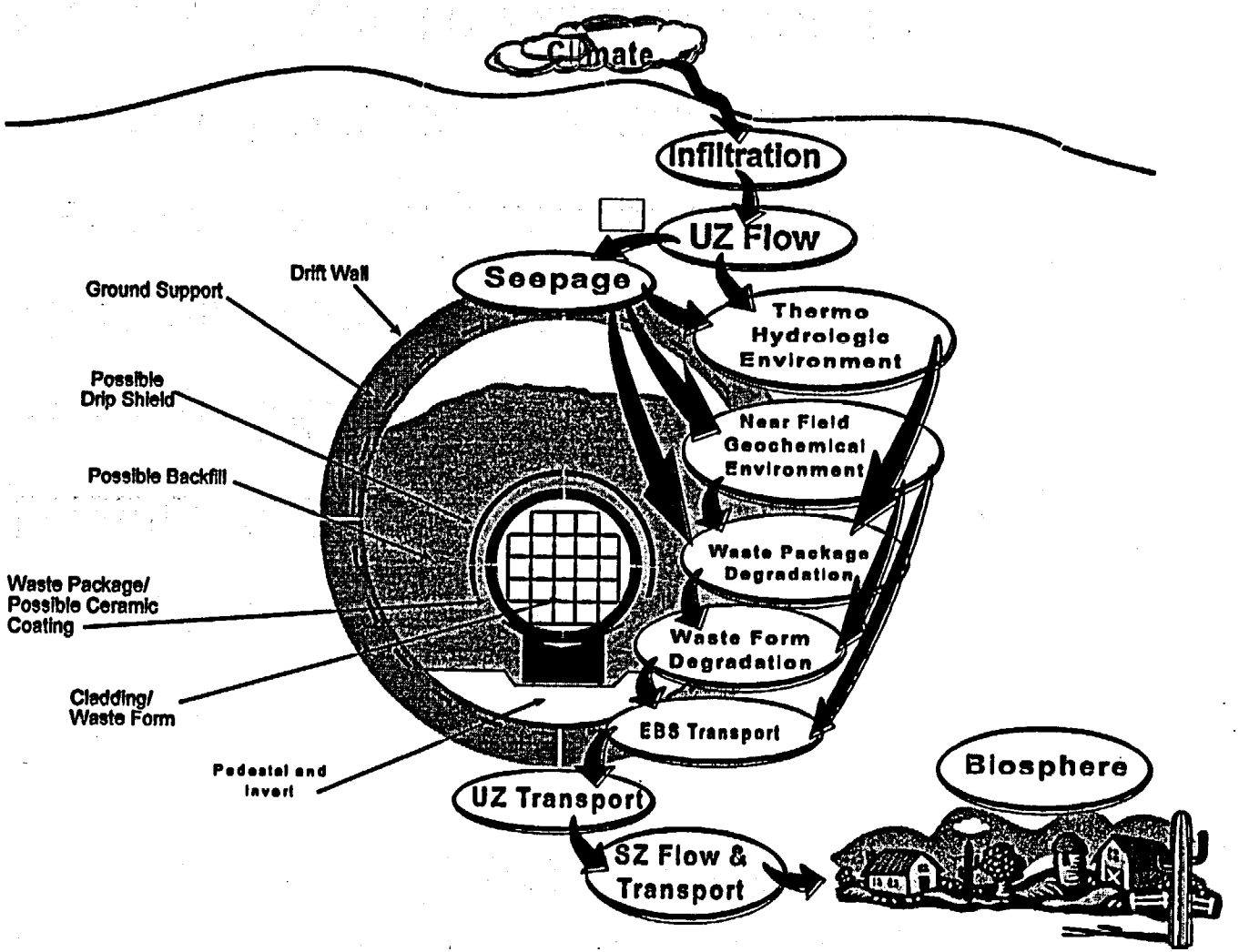
Sr. Technical Advisor

May 7, 1998

What is a TSPA "Base Case?"

- The development and documentation of our estimate of the probable behavior ("base case") of the repository reference design
- This analysis includes performance predictions of the multiple barrier system over long periods of time

Components of Total System Performance Assessment



NUKWSTP2.CDR.129.6-6-97

Definition of Base Case - Key Features of Natural System

- Climate
- Infiltration
- Unsaturated zone flow
- Seepage into drifts
- Thermal-hydrologic environment
- Near-field geochemical environment
- Unsaturated zone flow and transport
- Saturated zone flow and transport
- Biosphere

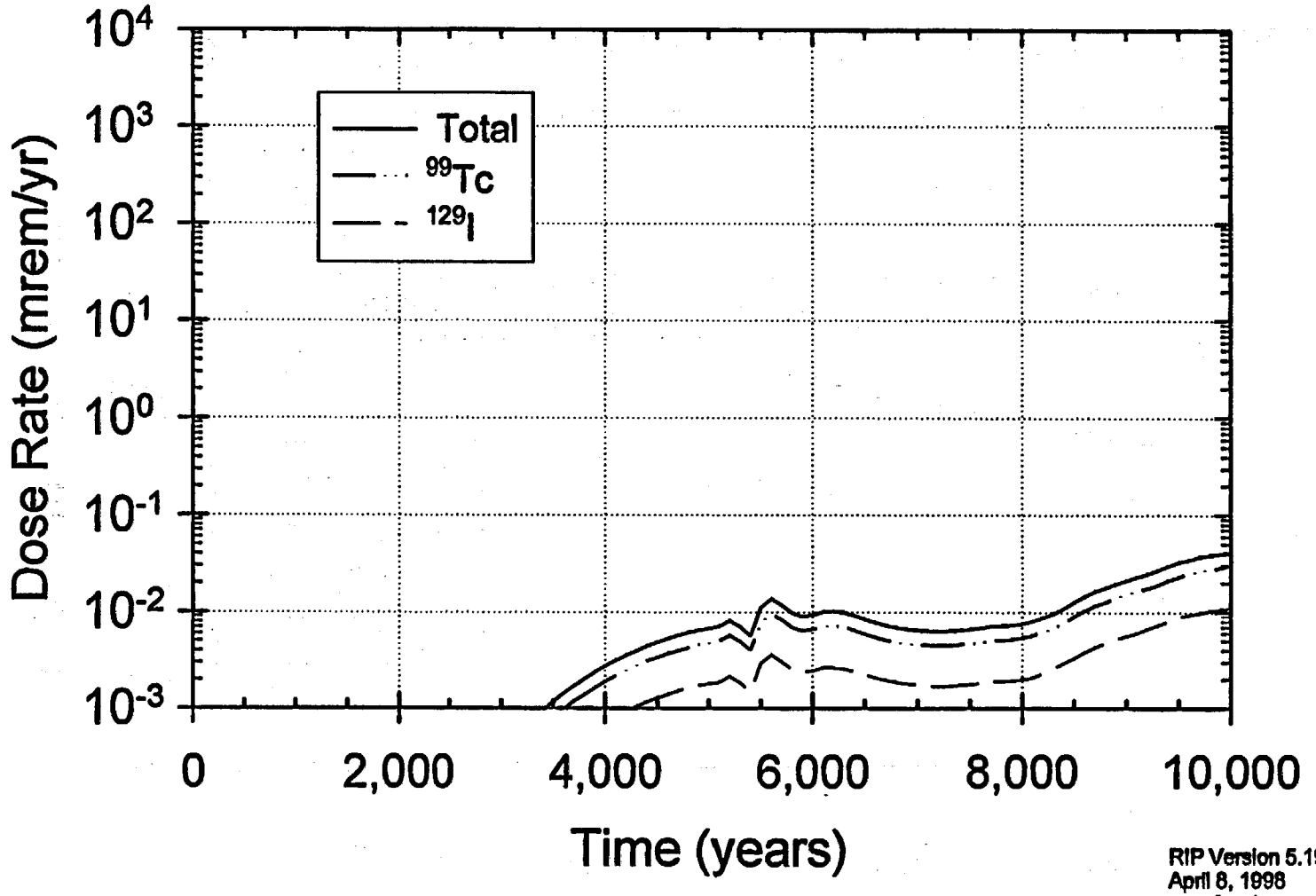
Definition of Base Case - Key Features of Engineered System

- Waste package degradation
- Cladding
- Waste form dissolution
- Radionuclide mobilization
- Engineered barrier system transport

Definition of Base Case - Key Features of TSPA-VA Design

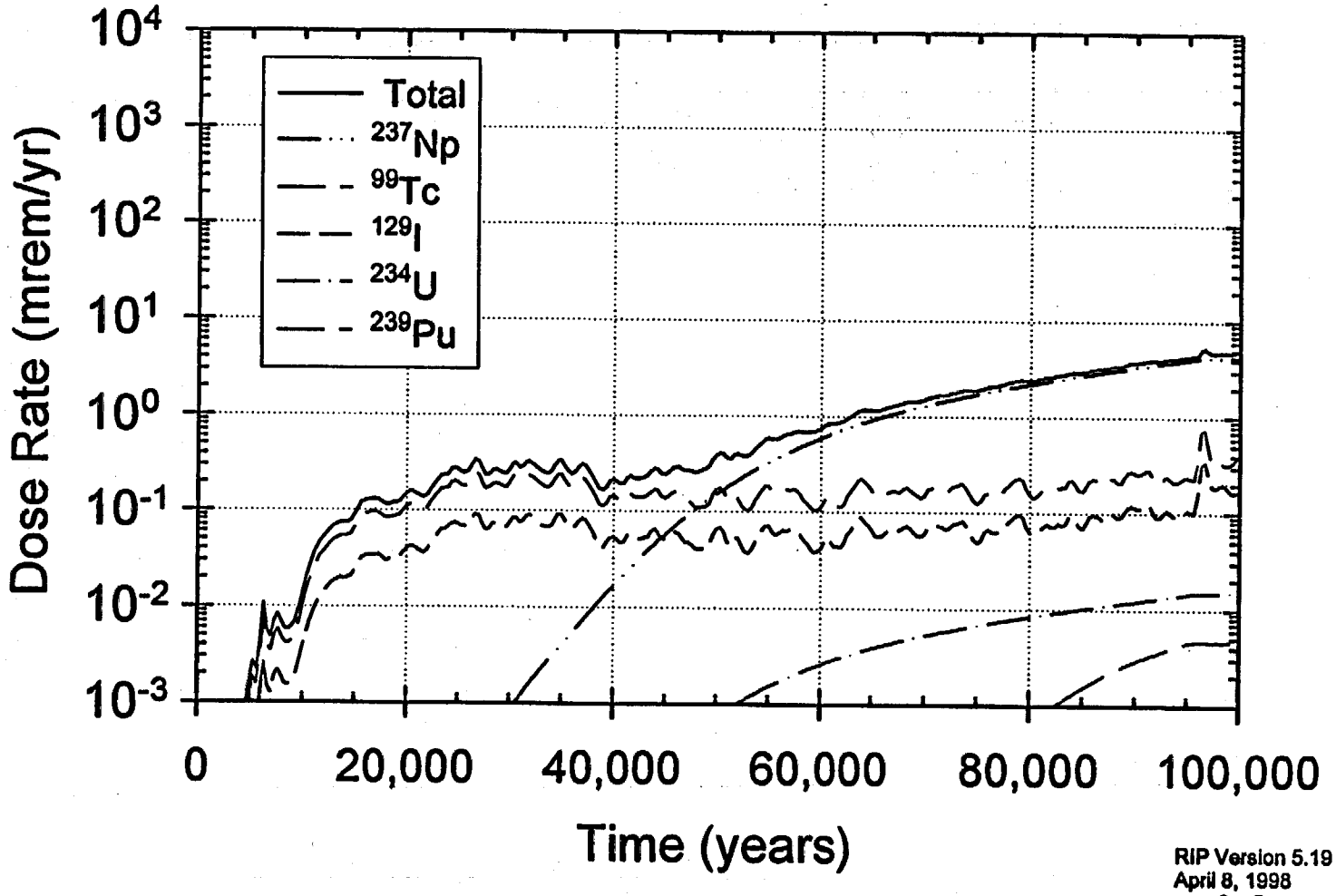
- **Repository**
  - 85 MTHM/acre
  - 70,000 MTHM total (includes commercial, DOE, Navy, and MOX fuel, and HLW)
- **Engineered barrier/emplacement drift**
  - 5.5 m diameter (20 cm concrete liner)
  - mild steel support for waste package
  - concrete invert
- **Waste package**
  - 21-PWR or 44-BWR commercial spent fuel packages
  - 10 cm mild steel outer barrier
  - 2 cm alloy C-22 inner barrier
  - 5 co-disposed HLW canisters/package

Expected-Value Base Case: 10,000-yr Dose-Rate History
Average Individual, All Pathways, at 20 km



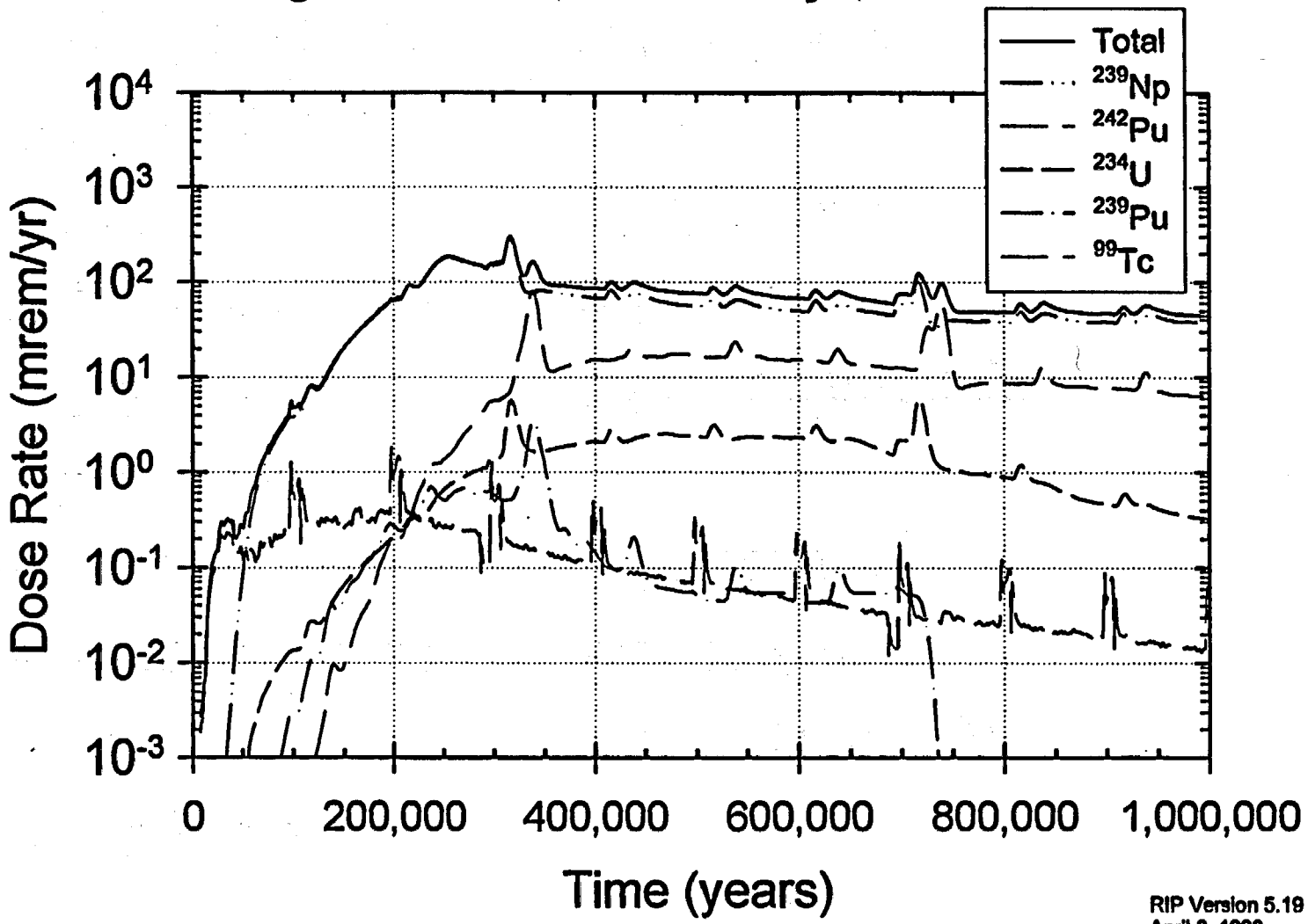
RIP Version 5.19
April 8, 1998
case0ee4

Expected-Value Base Case: 100,000-yr Dose-Rate History
Average Individual, All Pathways, at 20 km



RIP Version 5.19
April 8, 1998
case0ee5

Expected-Value Base Case: 1,000,000-yr Dose-Rate History
Average Individual, All Pathways, at 20 km



RIP Version 5.19
April 8, 1998
case0ee6

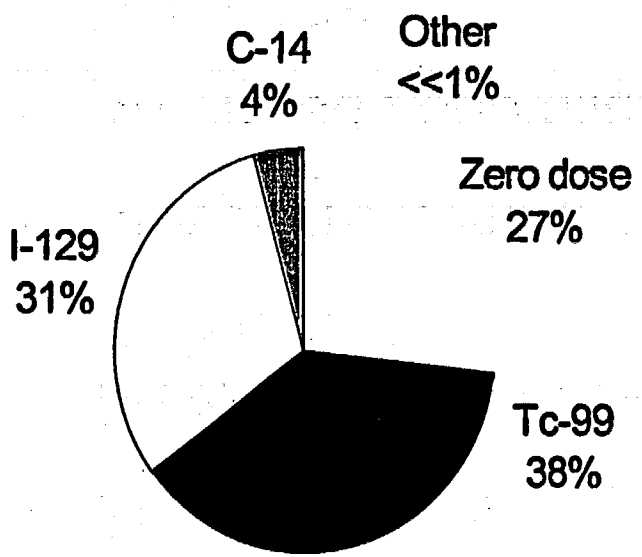
Review of Results - Peak Dose Values as a Function of Time

- **10,000 years**
  - 5% — 3.3×10^{-4} mrem/yr
  - expected value — 0.04 mrem/yr
  - 95% — 0.8 mrem/yr
- **100,000 years**
  - 5% — 1.98×10^{-4} mrem/yr
  - expected value — 5.3 mrem/yr
  - 95% — 187.7 mrem/yr
- **1,000,000 years**
  - expected value — 300 mrem/yr

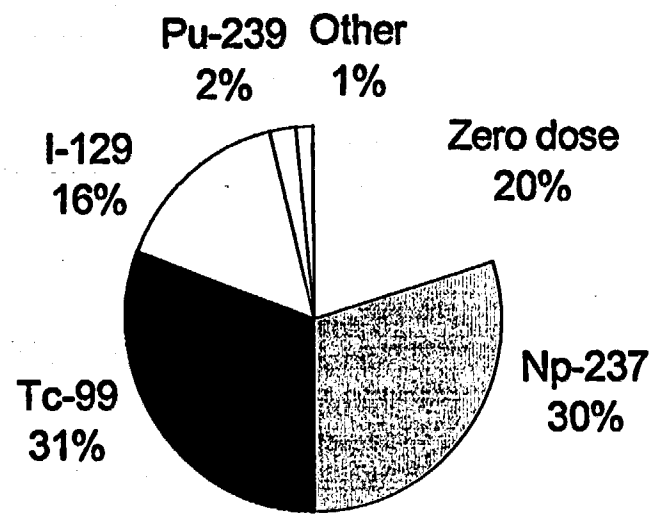
5% and 95% values based on multiple realizations

Radionuclide Contributions to Average Dose Rates in 100 Calculations

10,000-Year Period



100,000-Year Period



Most Important Parameters 10,000-Yr Peak Dose

- Fraction of waste packages contacted by seeps
- Mean corrosion rate of the corrosion-resistant material, C-22
- Biosphere dose-conversion factors
- Saturated zone dilution factor
- Seep flow rate

Most Important Parameters 100,000-Yr Peak Dose

- Fraction of waste packages contacted by seeps
- Mean corrosion rate of the corrosion-resistant material, C-22
- Variability in corrosion rate of the corrosion-resistant material, C-22
- Seep flow rate
- Saturated zone dilution factor
- Biosphere dose-conversion factors

Discussion:
Where Do We go From Here

- TSPA-VA is a tool that will be continually refined and applied throughout the SR/LA
- TSPA-VA results and uncertainty analyses will help prioritize the work needed for a defensible SR/LA
- **Defensibility**
  - Major progress toward gaining internal consensus on data and models used in analyses
  - Defensibility of analyses for SR/LA is the responsibility of every manager and organization

Discussion:
Where Do We go From Here
(continued)

- **Traceability and transparency**
  - Clarity of assumptions and analyses, and ability to track data and information through the analyses is critical to the success of the SR/LA
  - The QA exceptions effort that will be completed this summer will help to identify where work is needed for the SR/LA
- **Acceptability**
  - Presentations to NRC (March 17-19) and NWTRB (April 23-24) have been made in an effort to ensure "no surprises" during the review of the VA product
  - NRC stated that the TSPA-VA "has made significant progress to date in completing an imposing task"
  - "NRC's independent analysis . . . has identified a number of positive attributes of DOE's approach and a number of questions . . ."

Important Intermediate Results

- Commercial spent fuel dominates the source term for releases
  - much larger contribution than DOE spent fuel and HLW
- Advective releases are approximately an order of magnitude greater than diffusive releases
- To the extent that lateral flow is important to unsaturated zone transport, retardation due to sorption in the Calico Hills is less likely

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Director's Program Review
Yucca Mountain Project
Performance Measurement

PRESENTED TO

Lake Barrett, Acting Director

OCRWM

PRESENTED BY

Wayne N. Kozai, Acting Director

Office of Project Control

May 7, 1998

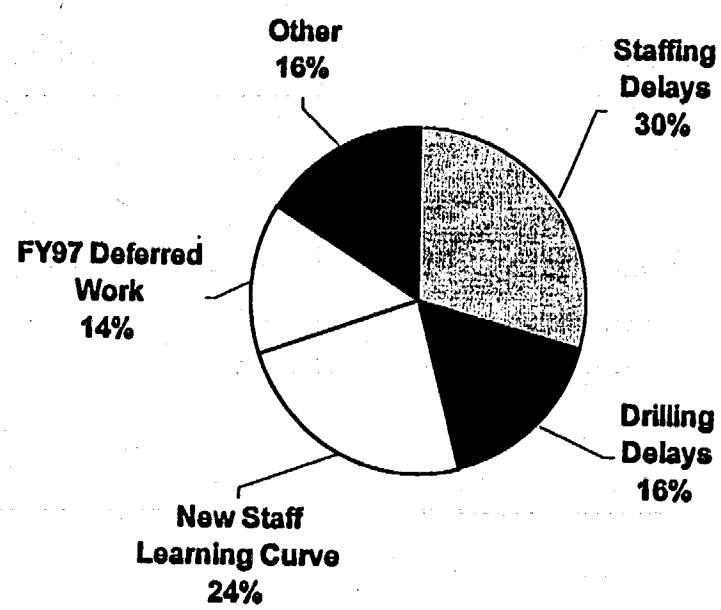
COMPARISON OF ORIGINAL CONTRACTOR BUDGET BASELINE (CBB) TO CURRENT CBB ($000)

CBB as of 9/30/97 (FY96 and FY97):			$ 489,090
CBB (FY98) as of 1/31/98:			298,141
Changes Since Last DPR:			
CR 98/008	Improve Project Processes in Document Preparation	$ 38	
CR 98/012	HLNW Conference	79	
CR 98/013	WT-24 Borehole Testing	160	
CR 98/014	East/West Drift Hydrologic Report	42	
			319
CBB as of 3/31/98			$787,550

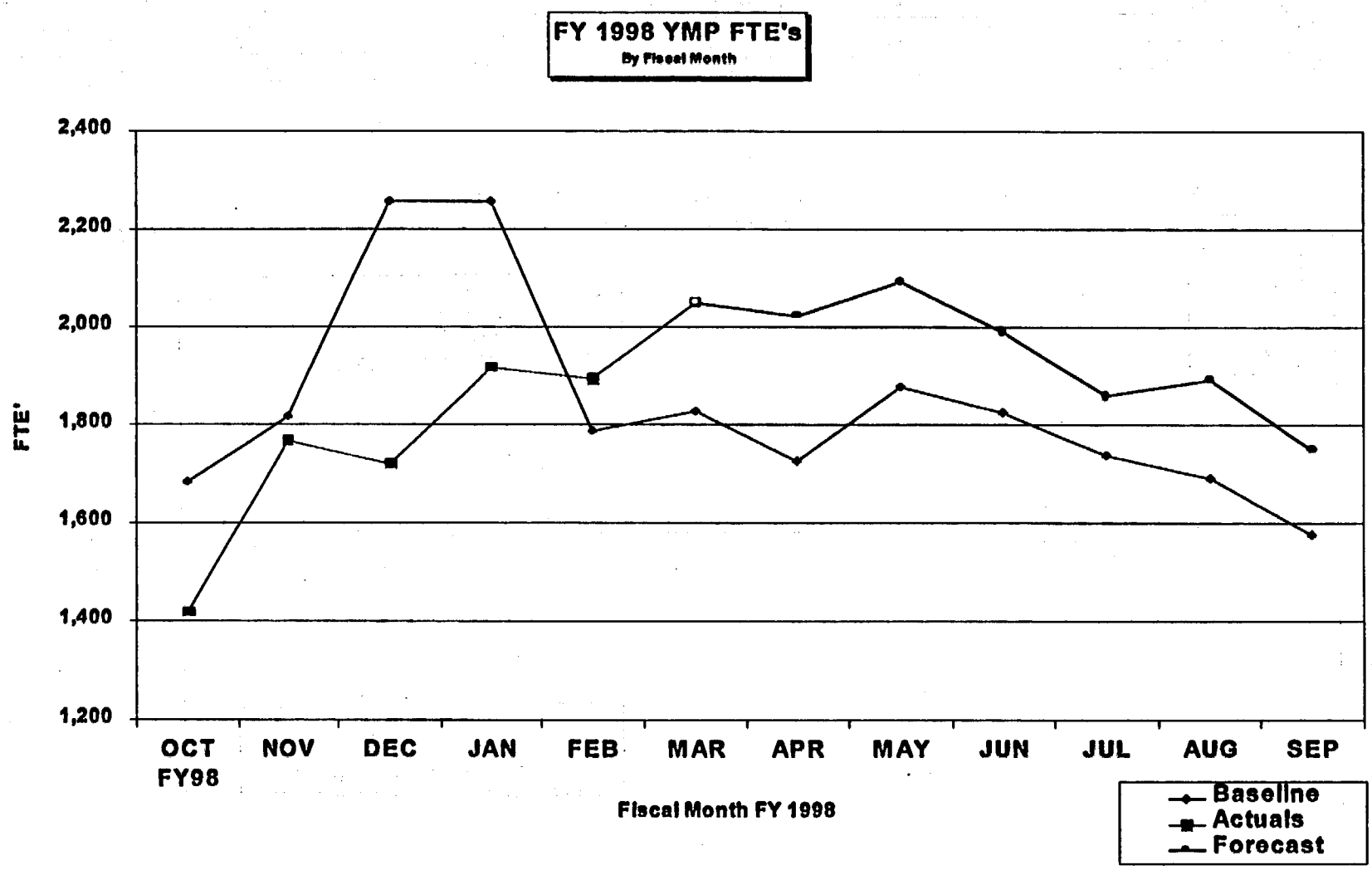
FY96/97/98 Project Budget Performance

	(a)	(b)	(c)	(d)=(b-a)	(e)=(b-c)	(f)	(g)	(h)=(f-g)
	FY 96/97/98 (ITD)					(FY96/97/98 At Completion)		
	Budgeted Cost		Actual Cost	Variances				
WBS	Work Scheduled BCWS	Work Performed BCWP	Work Performed ACWP	Schedule SV	Cost CV	Budget at Complete BAC	Latest Revised Estimate LRE	Variance at Complete VAC
1.2.1	31,607	29,687	27,803	(1,920)	1,884	41,702	40,591	1,111
1.2.2	37,778	36,849	34,152	(929)	2,697	48,928	48,607	321
1.2.3	147,984	145,316	143,424	(2,668)	1,892	178,697	178,526	171
1.2.4	33,391	31,918	31,307	(1,473)	611	43,906	42,201	1,705
1.2.5	51,497	49,632	48,624	(1,865)	1,008	65,739	65,792	(53)
1.2.6	147,803	146,505	158,986	(1,298)	(12,481)	173,547	191,416	(17,869)
1.2.7	30,695	30,365	26,958	(330)	3,407	36,592	32,462	4,130
1.2.8	49,417	49,434	49,949	17	(515)	61,757	63,490	(1,733)
1.2.9	42,391	42,380	38,746	(11)	3,634	52,697	51,295	1,402
1.2.12	32,625	32,621	35,549	(4)	(2,928)	40,391	45,727	(5,336)
1.2.14	8,653	8,649	8,432	(4)	217	10,703	10,710	(7)
1.2.15	27,409	27,411	26,182	2	1,229	32,891	31,753	1,138
	641,250	630,767	630,112	(10,483)	655	787,550	802,570	(15,020)

FY96/97/98 YMP Schedule Variance
($000)



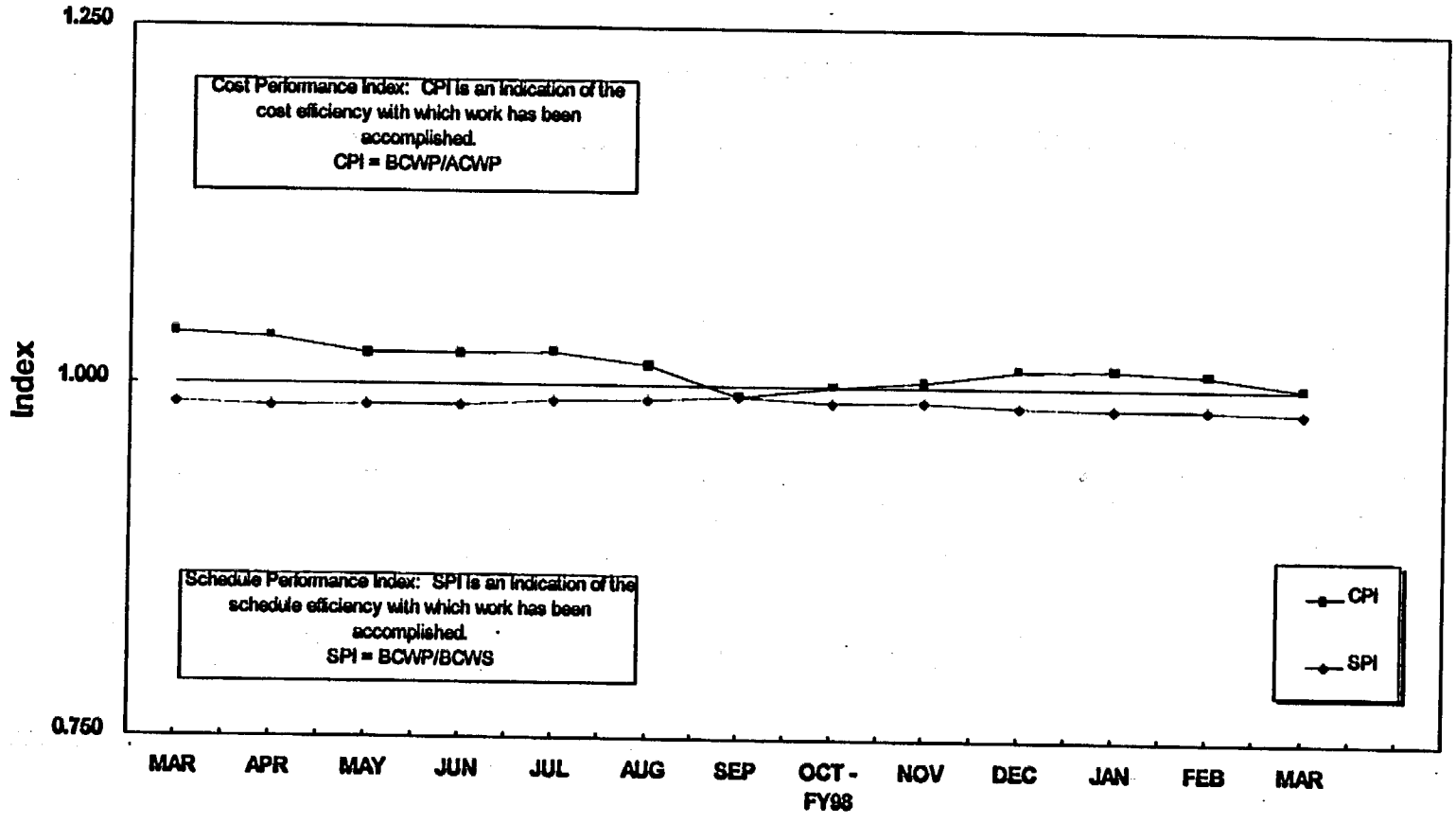
Staffing Delays	3,100
Drilling Delays	1,700
New Staff Learning Curve	2,500
FY97 Deferred Work	1,500
Other	1,683
Total	10,483



Fiscal Mo.	CT FY9	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP
Baseline	1,684	1,817	2,258	2,257	1,788	1,828	1,726	1,876	1,825	1,739	1,692	1,575
Actuals	1,417	1,767	1,720	1,915	1,895	2,050						
Forecast							2,023	2,092	1,990	1,860	1,892	1,750

PERFORMANCE ANALYSIS FY 1996/1997/1998

WBS 1.2, CPI & SPI OVERVIEW



	MAR	APR	MAY	JUN	JUL	AUG	SEP	CT - FY9	NOV	DEC	JAN	FEB	MAR
CPI	1.035	1.032	1.022	1.022	1.023	1.014	0.993	1.000	1.005	1.012	1.014	1.011	1.001
SPI	0.986	0.985	0.985	0.985	0.989	0.989	0.993	0.988	0.989	0.987	0.986	0.986	0.984

Variance Analysis Report

WBS 1.2.1 Systems Engineering

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
159,850	157,930	156,046	-1,920

SV: Development of SDDs (-612)
Fissile Material Disposition (-751)

Cause: Delay in staffing at planned levels and impact from significant SDD learning curve.

Impact: No impact on VA.
Recovery beginning now with significant recovery by July/August 1998

Variance Analysis Report

WBS 1.2.1 Systems Engineering

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
159,850	157,930	156,046	-1,920

Corrective Action:

6 of 17 Personnel Requisitions have been filled.
12 TDY and 3 Home Office support are being utilized.

Variance Analysis Report

WBS 1.2.2 Waste Package

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
150,669	149,740	147,043	-929

SV: -760 is deferred from FY97. FY98 contribution is 170.

Cause: Over $557 K is attributable to subcontract and ODC delays. The remainder ($372 K) is due to insufficient staffing.

Impact: No impact on deliverables and major feeds. Slip non-critical work.

Corrective Action: Focus current resources on early milestones and feeds. Continue to increase staff and home office support and continue limited EWW.

Variance Analysis Report

WBS 1.2.3 Site Investigations

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	BAC	LRE	VAC
691,497	688,829	686,937	-2,668	918,423	917,992	431

SV: -2,668 (.4%) (-1,641 due to current year; -1027 due to deferred/deleted scope in prior years. All but 140 of the deferred work has been recovered.)

Drilling and Associated Delays (-1,700)

Cause: Drilling problems/delays at WT-24 and SD-6 and consequent delay in associated data collection and analysis contribute to the variance.

Impact: Drilling activities will remain behind schedule. Some associated level 4 milestones will be delayed; these are not associated with VA products.

Corrective Action: Recovery plans include working a 4 shift/day instead of a 3 shift/day, improve field drilling performance with rental equipment, and personnel changes.

Variance Analysis Report

WBS 1.2.3 Site Investigations

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	BAC	LRE	VAC
691,497	688,829	686,937	-2,668	918,423	917,992	431

SV: -2,668 (.4%) (-1,641 due to current year; -1027 due to deferred/deleted scope in prior years. All but 140 of the deferred work has been recovered.)

Waste Package Environment (Near-Field Testing) (-300)

Cause: Staffing deficiencies have contributed to schedule delays.

Impact: Although the near-field testing is behind schedule, no deliverables are expected to be impacted in the FY 1998 or FY 1999 schedule.

Corrective Action: Schedule is anticipated to recover in September. An accelerated effort will begin in June through staffing of new hires and summer students.

Variance Analysis Report

WBS 1.2.3 Site Investigations

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	BAC	LRE	VAC
691,497	688,829	686,937	-2,668	918,423	917,992	431

SV: -2,668 (.4%) (-1,641 due to current year; -1027 due to deferred/deleted scope in prior years. All but 140 of the deferred work has been recovered.)

Geophysical Analyses for ECRB (-250)

Cause: A new approach to conducting the activity initially resulted in slow progress.

Impact: Schedule float between this activity and successor activities is reduced.

Corrective Action: Work will progress at an accelerated pace. Several tasks under the old approach are no longer required, therefore, deliverable dates are anticipated to be met.

Variance Analysis Report

WBS 1.2.3 Site Investigations

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	BAC	LRE	VAC
691,497	688,829	686,937	-2,668	918,423	917,992	431

SV: -2,668 (.4%) (-1,641 due to current year; -1027 due to deferred/deleted scope in prior years. All but 140 of the deferred work has been recovered.)

Progress Report Site Input/Review (-120)

Cause: DOE has directed changes to the schedule and the content for Progress Report 18. The input date has been extended and the length of report reduced.

Impact: Because the change request (M&O-98-25) is in progress to represent the new direction and related Level 3 milestones are being revised, there will be no impact to other milestones.

Corrective Action: A change request is in progress to represent the new direction and change in schedule.

Variance Analysis Report

WBS 1.2.3 Site Investigations

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	BAC	LRE	VAC
691,497	688,829	686,937	-2,668	918,423	917,992	431

VAC: 431 (.05%)

Cause: The VAC is not significant.

Impact: None.

Corrective Action: No action required.

Variance Analysis Report

WBS 1.2.4 Repository

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
148,872	147,399	146,788	-1,473

SV: There is a -$276K associated with 1.2.4.2, -$624K associated with 1.2.4.6 of with -$305 is caused by the Deferred Work from FY97, and -$572K associated with 1.2.4.7.

Cause: Work ramp-up for Waste Handling System Design and Site Generated Radiological Waste Systems was delayed to align with the SDD development delays. Planned work on the PISA and ICDs has not taken place as planned.

Variance Analysis Report

WBS 1.2.4 Repository

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
148,872	147,399	146,788	-1,473

Impact: There are no impacts or delays expected to level 3 deliverables or milestones associated with these variances.

Corrective Action: Look for ways to improve performance in the areas of Waste Handling System Design, Site-Generated Radiological Waste Systems and VA Design. Much of the SV will be mitigated with the approval of CR (M&O-98-005) which will address work-scope and budget changes from PISA to WDLA

Variance Analysis Report

WBS 1.2.5 Regulatory

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
227,297	225,423	224,424	-1,865

SV:

Cause: 1.2.5.2 Amount of effort on placement of VA documents was reduced so that PEMP 11-4 related work could be performed, simultaneously (-$350K)

CR for PISA to WDLA not yet approved (-$150K)

1.2.5.3 Behind in staffing (-$200K)

1.2.5.4 Late Implementation of QA CR (-$250K)

Late finalization of TSPA-VA base case (-$400K)

1.2.5.7 Incorrect data entry

1.2.5.10 Transition in subcontractor staffing (-$250K)

Variance Analysis Report

WBS 1.2.5 Regulatory

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
227,297	225,423	224,424	-1,865

SV:

Impact: 1.2.5.2 Carrying variance for PISA until CR completed.

1.2.5.3 Limited staffing caused a backlog in GENIIS data input.

1.2.5.4 All QA milestones will be met for the Technical Basis Document.

The part of the SV from final decisions on definition of base case are now on schedule and lost schedule is being recovered.

1.2.5.7 None

1.2.5.10 Backlog in records

Variance Analysis Report

WBS 1.2.5 Regulatory

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV
227,297	225,423	224,424	-1,865

SV:

Corrective Action:

1.2.5.2 Complete PISA/WDLA CR.

Following successful completion of PEMP 11-4, focus on baselined activities in WBIS/TOC. Deliverables are expected to be completed on schedule

1.2.5.3 Staff is being put in place for TDM; 13 new staff are in place & ~10 additional requisitions are out. Expect to be staffed in summer

1.2.5.4 With the revised definition of the TSPA-VA base case work has begun on the TSPA analyses for the EIS alternatives using off site staff to recover schedule

1.2.5.7 Correct data next month

1.2.5.10

Variance Analysis Report

WBS 1.2.6 Exploratory Studies Facility

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	SV	CV	BAC	LRE	VAC
464,889	463,591	476,072	-1,298	-12,481	586,122	603,247	-17,125

SV: (Cause) The SV is mostly due to Drift Scale Test activities which are reported behind schedule as a result of details added in mid - FY97. Other impacts include late start on additional designs per TER's, and Title III Evaluation Reports.

CV: (Cause) The unfavorable cost variance is the result of FY96 and FY97 activities exceeding budget. FY96 activities exceeded budget due to ESF tunneling costs that were greater than planned as a result of extremely poor ground conditions requiring additional steel sets, constructor management time and excavation time. FY97 activities exceeded budget mostly as a result of unplanned constructor's insurance/bond costs and SIIS charges.

VAC: The unfavorable VAC is mostly the result of FY96 and FY97 activities as noted above. Also, FY98 constructor staff levels are greater than budgeted.

Variance Analysis Report

WBS 1.2.6 Exploratory Studies Facility

Multi-Year Cumulative to Date Schedule and At Complete

Impact: No impacts to any future milestones are expected.

Corrective Action: No corrective action required for SV and CV. Actions to reduce the year-end variance are currently being evaluated.

Variance Analysis Report

WBS 1.2.8 Environmental, Safety and Health

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	CV	BAC	LRE	VAC
99,367	99,384	99,899	-515	161,113	162,846	-1,733

CV:

FY96 - $1.1M positive variance due to NTS services budgeted in 1.2.8, but costed in 1.2.7

FY97 - $1.3M negative variance due to K/PB support for ongoing activities after TBM holeout

FY98 - $0.3M negative variance primarily due to EIS "Appendix E" early start activities while waiting change request approval, and K/PB overrun

Variance Analysis Report

WBS 1.2.8 Environmental, Safety and Health

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	CV	BAC	LRE	VAC
99,367	99,384	99,899	-515	161,113	162,846	-1,733

VAC:

In addition to cumulative cost variance described above, forecast includes an additional $0.6M in late billing FY97 commitments and deferred work, $0.4M estimated labor forecast above budgeted rates, and $0.2M in K/PB costs above original budget.

Cause:

See above

Variance Analysis Report

WBS 1.2.8 Environmental, Safety and Health

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	CV	BAC	LRE	VAC
99,367	99,384	99,899	-515	161,113	162,846	-1,733

Impact:

1. No multiyear impact regarding offsetting FY96 and FY97 variances
2. FY98 EIS activities remain overrun until change request approval
3. Negative FY98 labor variance may be offset by other third level WBS currently experiencing positive labor variances.

Corrective Action:

None

Variance Analysis Report

WBS 1.2.12 Information Management

Multi-Year Cumulative to Date Schedule and At Complete

BCWS	BCWP	ACWP	CV	BAC	LRE	VAC
67,816	67,812	70,740	-2,928	146,783	152,119	-5,336

CV: -2,928

VAC: -5,336

Cause: 1) An overrun of $-3.5M in the M&O Computer System Upgrade.
2) An overrun of $-1.0M in the Kenrob Computer System Upgrade.
3) An overrun of $-1.315M in Kenrob carryover LRE as directed by the DOE.
4) Underruns in other Information Management areas.

Impact: None. The contracts have been funded to cover the upgrade.

Corrective Action: None. The PMB was not updated per DOE direction.

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
DIRECTOR'S PROGRAM REVIEW

WAST Project Overview

PRESENTED TO

LAKE BARRETT, Acting Director

OCRWM

PRESENTED BY

Dwight Shelor, Acting Director
Office of Waste Acceptance, Storage and Transportation

May 7, 1998

PRIORITY ACTIVITIES

- **Waste Acceptance and Transportation**
  - RFP for Waste Acceptance and Transportation Services
  - Standard Contract Management
  - Section 180(c) Development
- **Interim Storage Related Activities**
  - Phase I Centralized Interim Storage Facility (CISF) TSAR - support of NRC Review
  - Dry Transfer System (DTS) TSAR - support of NRC review
  - Actinide-only Burnup Credit (BUC) Topical Report Rev.1 - support of Nuclear Regulatory Commission (NRC) Review

Shelor
5/6/98

CHANGES SINCE LAST DPR
February - March

- BCP-03-98-0001 - New Work Scope and Budget to Support CISF TSAR Licensing (+$306K).

Shelor
5/6/98

ACCOMPLISHMENTS
March - April

- Preparing responses to NRC request for additional information on the CISF TSAR
  - Request received on March 9, 1998
  - Met with NRC staff on April 14, 1998
  - Response planned for early June.

- Authorized the redistribution of funds to support testing of the DTS prototype at Idaho National Engineering Laboratory in June 1998.

- Developing responses to NRC's April 3, 1998 request for additional information on the Actinide-Only BUC Topical Report
  - Met with NRC Staff on April 28, 1998.

Shelor
5/6/98

ACCOMPLISHMENTS (Cont.)
March - April

- Issued the NWPA Section 180(c) Notice of Revised Proposed Policy and Procedures on April 30, 1998.
- Closed public comment period on Revised Draft RFP for Acquisition of Waste Acceptance and Transportation Services on April 13, 1998
  - Thirty-two comment letters placed in the DOE Public Reading Room.
- WAST Project Summary Schedule (PSS) and WAST Multi-Year Cost and Schedule Baseline Rev. 5 prepared and in concurrence.
- Issued cooperative agreement award to Western Interstate Energy Board on April 9, 1998 (New 5-year contract).

Shelor
5/6/98

ACCOMPLISHMENTS (Cont.)
March - April

- Provided presentations/briefings to:
  - Senior Executive Forum (Transportation), DOE-HQ on March 5
  - Department of Energy Indian Issues Points-of-Contact Meeting, Santa Fe, NM on March 17-18
  - National Congress of American Indians, High-Level Radioactive Waste Committee, Reno, NV on April 7
  - National Radiological Emergency Preparedness Conference, Hyannis, MA on April 27, 1998.

Shelor
5/6/98

LEVEL 1 MILESTONE STATUS

WBS Level	Milestone (ID Number)	BCCB Level	Current Baseline Date	Actual (A)/ Forecast Date
3.1.1.1	Issue Final RFP for Acquisition of RSC WA&T Services (WT311M7004)	1	TBD	8/01
3.1.4	Issue NWPA Sec. 180 (c) Notice of Revised Proposed Policy and Procedures (3140M005)	1	2/98	4/98 (A)
3.2.2.1	Receive Phase I CISF/TSAR Safety Assessment Report from the NRC (WI322M8008)	1*	9/98	11/98
3.2.2.3	Receive DTS TSAR Safety Assessment Report from the NRC (WI322M8001)	1*	4/98	11/98
3.2.4.1	Receive Actinide-Only BUC Safety Assessment Report from the NRC (WI324M8001)	1*	11/97	4/00

* To be changed to Level 2 milestones in the next revision of the baseline

Shelor
5/6/98

LEVEL 2 MILESTONE STATUS

WBS Level	Milestone (ID Number)	BCCB Level	Current Baseline Date	Actual (A)/ Forecast Date
3.1.1.1	Complete Final RFP for Acquisition of RSC WA&T Services (WT314M700)	2	TBD	7/01
3.1.4	Complete NWPA Sec. 180 (c) Notice of Revised Proposed Policy and Procedures (3140M007)	2	12/97	2/98 (A)
3.1.4	Complete Draft Application Packages for NWPA Sec. 180 (c) Grants (3140M017)	2	6/98	9/99
3.2.1.1	Complete Conceptual Design for Non-Site-Specific Phase II CISF (WI321M8001)	2	8/98	To Be Deleted
3.2.1.1	Approve CISF Design Requirements Document Rev. 2 by POBCCB (WI321M8001)	2	12/97	9/98

Shelor
5/6/98

LEVEL 2 MILESTONE STATUS

WBS Level	Milestone (ID Number)	BCCB* Level	Current Baseline Date	Actual (A)/ Forecast Date
3.3.1.2	Complete FY 1999 WAST Annual Work Plan (WP331M8001)	2	9/98	9/98
3.3.1.3	Complete FY 2000 OMB Submittal for the WAST Project (WP331M8002)	2	8/98	8/98

Shelor
5/6/98

DPR ACTION ITEMS

- None.

Shelor
5/6/98

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

DIRECTOR'S PROGRAM REVIEW
WAST PROJECT
Performance Measurement

PRESENTED TO

LAKE BARRETT, Acting Director OCRWM

PRESENTED BY

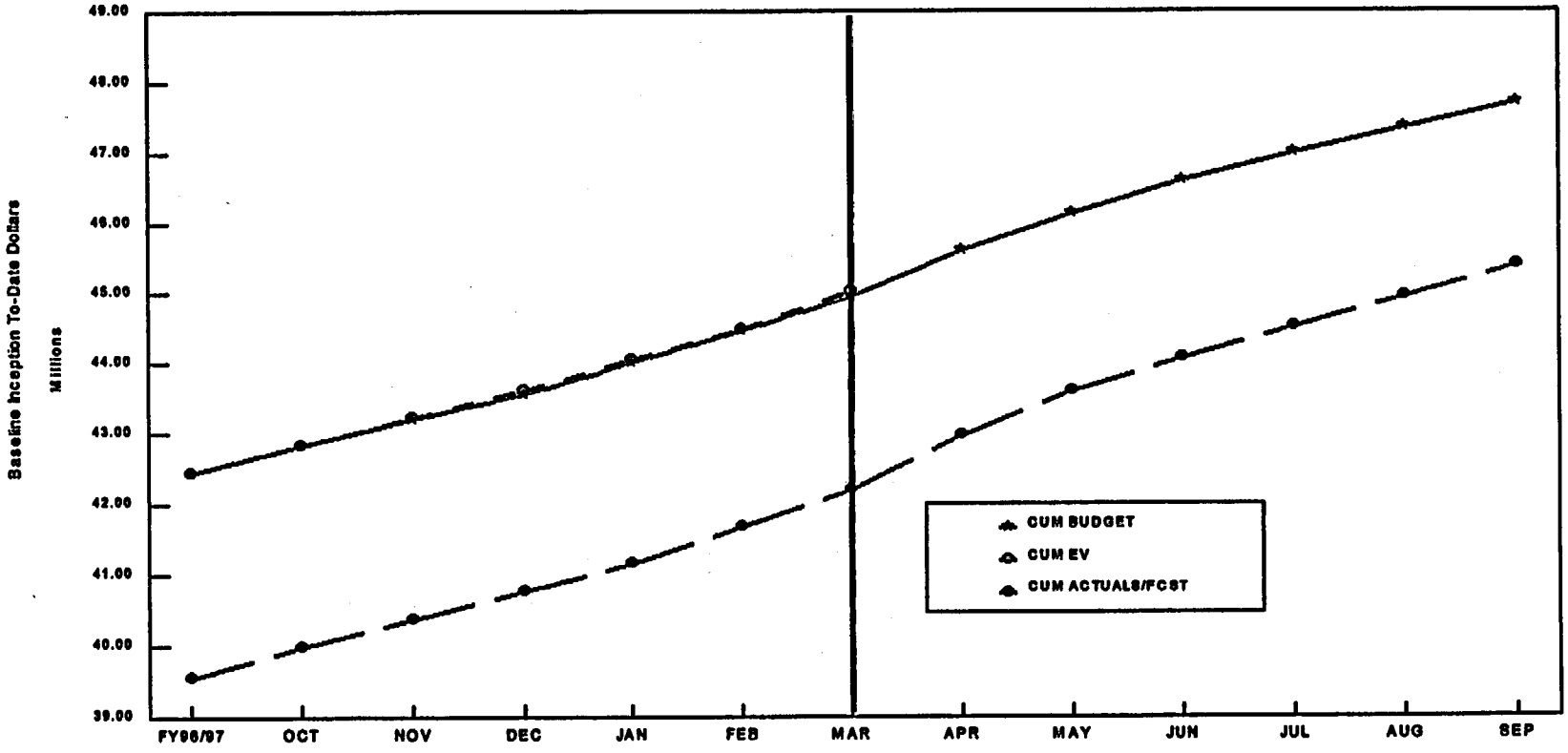
SYED BOKHARI,
Senior Project Management Integrator
Office of Waste Acceptance,
Storage and Transportation

May 7, 1998

COMPARISON OF ORIGINAL CONTRACTOR BUDGET BASELINE (CBB) TO CURRENT CBB ($K)

CBB(as of 9/30/97):	($)42,458
CBB(as of 3/31/98):	($)47,612
Changes(since last DPR):	($) 306
CBB(current CBB):	($)47,918

FY 1996/98 3.0 WAST FYTD EV Chart



	FY96/97	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	FY99
MONTHLY BUD	42457	394	360	354	458	441	500	661	530	474	399	349	344	196
MONTHLY EV	42457	396	388	378	438	435	558							
MONTHLY ACT	39563	429	407	375	399	510	524							
MONTHLY FCST	39563	429	407	375	399	510	524	766	640	468	451	424	464	196
SCHED VAR	0	2	28	24	-20	-6	58							
COST VAR	2894	-33	-19	3	39	-75	34	0	0	0	0	0	0	0
CUM BUDGET	42457	42851	43211	43565	44023	44464	44964	45625	46155	46629	47028	47377	47721	47917
CUM EV	42457	42853	43241	43619	44057	44492	45050							
CUM ACTUAL	39563	39992	40399	40774	41173	41683	42207							
CUM FCST	39563	39992	40399	40774	41173	41683	42207	42973	43613	44081	44532	44956	45420	45616
ITD SV	0	2	30	54	34	28	86							
ITD CV	2894	2861	2842	2845	2884	2809	2843	0	0	0	0	0	0	0

5/7/98

U.S. Dep[illegible]nt of Energy
Cost Performance Report -Work Breakdown Structure (Format 1)
FY96-98 Month Ending March 1998

1. Title	2. Reporting Period	3. Dollars Expressed in Thousands Escalated
Waste Acceptance, Storage and Transportation	01-Mar-1998 to 31-Mar-1998	6. Report Date
4. Participant Name & Address	5. Cost Plan Date	
OCRWM		7. Manager's Signature

8. WBS Elements	January 1998					FY96-98 to Date					FY96-98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
3.1 Transportation	252	252	256	0	-4	16159	16159	14125	0	2034	17768	16147	1621
3.2 Interim Storage Facility (ISF)	150	208	182	58	26	8776	8862	8636	86	226	9540	9392	148
3.3 Project Management & Administration	98	98	86	0	12	4399	4399	4077	0	322	4979	4699	280
3.4 Casks/Canisters Tech. Development	0	0	0	0	0	15630	15630	15369	0	261	15630	15378	252
9. Contractor Budget Base	**500**	**558**	**524**	**58**	**34**	**44964**	**45050**	**42207**	**86**	**2843**	**47917**	**45616**	**2301**
10. Financial & Tech. Assistance													
11. Undistributed Budget											12500	12500	0
13. Fee											764	673	91
14. Baseline Adjustment/Contingency													0
15. Perf. Measurement Baseline											**61181**	**58789**	**2392**
16. Management Reserve											1190	1190	0
17. Other											2595	1993	602
18. Project Budget Base											**64966**	**61972**	**2994**

5/7/98

U.S. Department of Energy
Cost Performance Report - Work Breakdown Structure (Format 1)
FY98 Month Ending March 1998

1. Title Waste Acceptance, Storage and Transportation	2. Reporting Period 01-Mar-1998 to 31-Mar-1998	3. Dollars Expressed in Thousands Escalated
		6. Report Date
4. Participant Name & Address OCRWM	5. Cost Plan Date	7. Manager's Signature

8. WBS Elements	March 1998					FY-98 to Date					FY-98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
3.1 Transportation	252	252	208	0	44	1330	1330	1077	0	253	2939	2877	62
3.2 Interim Storage Facility (ISF)	150	208	181	58	27	642	728	748	86	-20	1406	1447	-41
3.3 Project Management & Administration	98	98	86	0	12	535	535	467	0	68	1115	1085	30
3.4 Casks/Canisters Tech. Development	0	0	0	0	0	0	0	0	0	0	0	0	0
9. Contractor Budget Base	**500**	**558**	**475**	**58**	**83**	**2507**	**2593**	**2292**	**86**	**301**	**5460**	**5409**	**51**
10. Financial & Tech. Assistance													
11. Undistributed Budget													0
13. Fee											211	195	16
14. Baseline Adjustment/Contingency													0
15. Perf. Measurement Baseline											**5671**	**5604**	**67**
16. Management Reserve											1143	1143	0
17. Other											3	3	0
18. Project Budget Base											**6817**	**6750**	**67**

WBS 3.0 WAST VARIANCE ANALYSIS

($000)

WBS	TITLE	BCWP	BCWS	ACWP	SV	CV
3.0	WAST	44,964	45,050	42,207	86	2,843

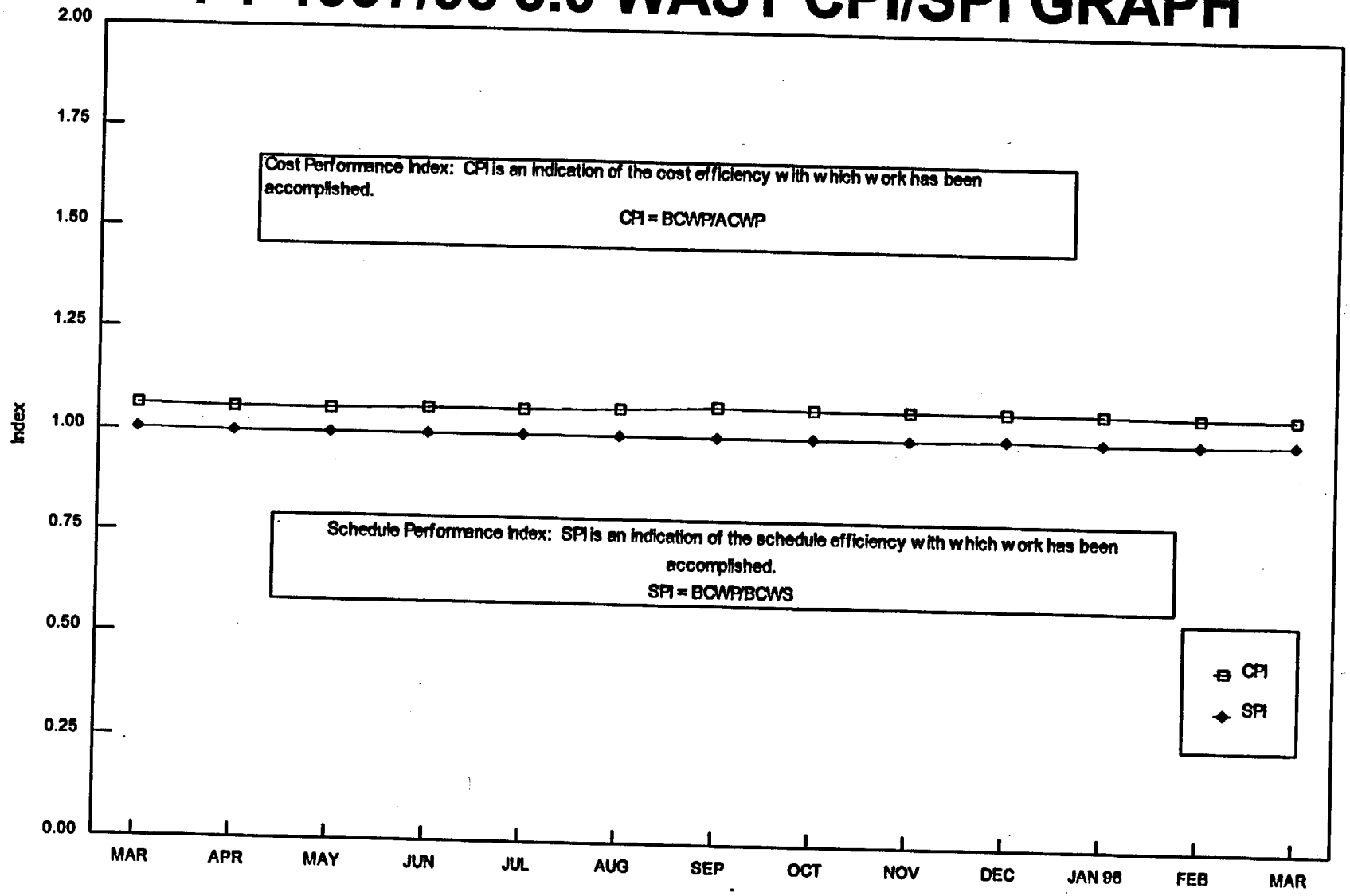
Schedule Variance: The CISF Enhanced-Handling Equipment Feasibility Assessment and Design Requirements Document (DRD) Rev 2 are being developed ahead of schedule.

Cost Variance: The cost variance reflects FY96 and FY97 work completed below budget.

Impact: None.

Corrective Action: None.

FY 1997/98 3.0 WAST CPI/SPI GRAPH



FY 1998	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	JAN 98	FEB	MAR
CPI	1.062	1.059	1.056	1.062	1.063	1.065	1.073	1.072	1.070	1.070	1.070	1.067	1.067
SPI	1.003	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.001	1.001	1.001	1.001	1.002

5/7/98

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

OFFICE OF QUALITY ASSURANCE

PRESENTED TO

LAKE BARRETT, ACTING DIRECTOR

PRESENTED BY

DONALD G. HORTON, Director
Office of Quality Assurance

May 7,1998

PRIORITY ACTIVITIES

- CONTINUE EFFORTS TO CONSOLIDATE OCRWM WIDE PROGRAM PROCEDURES WHERE PRACTICAL AND WITHOUT IMPACT TO VIABILITY ASSESSMENT

- SUPPORT THE RESOLUTION OF THE SIGNIFICANT CONDITIONS ADVERSE TO QUALITY, PARTICULARLY IN THE AREAS OF PROCUREMENT, SOFTWARE AND DATA QUALIFICATION

- CONTINUE TO SUPPORT PERFORMANCE ASSESSMENT AND OTHER CRITICAL ACTIVITIES FOR VIABILITY ASSESSMENT BY PROVIDING OQA SUPPORT PERSONNEL AS NEEDED

PRIORITY ACTIVITIES (Continued)

- PERFORM INSPECTIONS, SCHEDULED AUDITS AND QA MANAGEMENT ASSESSMENTS TO SUPPORT QA DELIVERABLE ACTIONS

- CONTINUE WORKING ON ACTION PLAN TO ADDRESS ISSUES RAISED AS A RESULT OF VISITS TO THE NATIONAL LABORATORIES AND THE USGS

CHANGES SINCE LAST DPR
February-March

NONE

ACCOMPLISHMENTS
March-April

- COMPLETED THE REORGANIZATION OF OQA AND ITS SUPPORT STAFF
  - Maximize Utilization of Resources (Skill-Mix)
  - Create Resource Flexibility
  - Improve Management Focus
  - Balance the Various Disciplines
- COMPLETED THE QA MANAGEMENT ASSESSMENTS AT LOS ALAMOS AND SANDIA NATIONAL LABORATORIES
- COMPLETED LEAD AUDITOR TRAINING COURSE FOR 18 INDIVIDUALS AT THE DOE RICHLAND, WASHINGTON OPERATIONS OFFICE

ACCOMPLISHMENTS
March-April 1998

• AUDITS	
– INTERNAL	3
– SUPPLIER	6
• SURVEILLANCES	7
• DEFICIENCIES	
– OPENED	47
– CLOSED	64
• QA DOCUMENTS	
– DEVELOPED	22 (1)
– REVIEWED	362 (2)
• INSPECTION AND TEST REPORTS	
– INSPECTIONS	42
– TEST REPORTS	178

(1) - INCLUDES DOCUMENT REVISIONS
(2) - INCLUDES DESIGN DOCUMENTS, DATA PACKAGES, P.O. TEST PLANS, JOB PACKAGES, SCIENTIFIC NOTEBOOKS, ETC.

MAJOR FY98 DELIVERABLES
(LEVEL 1 AND SUPPORTING LEVEL 2)

DELIVERABLE	LEVEL	BASELINE DATE	FORECAST
QA MANAGEMENT ASSESSMENT	1	09-30-98	09-30-98
FY97 QA PROGRAM ANNUAL SUMMARY REPORT	2	03-01-98	Complete 01/13/98
PERFORM REGULATORY REQUIRED INTERNAL QA PROGRAM COMPLIANCE BASED AUDITS	2	09-30-98	09-30-98
PERFORM REGULATORY REQUIRED SUPPLIER AUDITS	2	09-30-98	09-30-98

DPR ACTION ITEMS

- NONE

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

OFFICE OF QUALITY ASSURANCE

Performance Measurement

PRESENTED TO

LAKE BARRETT, ACTING DIRECTOR

PRESENTED BY

DONALD G. HORTON, Director
Office of Quality Assurance

May 7, 1998

COMPARISON OF ORIGINAL CONTRACTOR BUDGET BASELINE (CBB) TO CURRENT CBB ($K)

CBB:	($)34,016
CBB(FY98):	($) 9,629
Changes(FY98):	($) -0-
CBB(current CBB):	($)43,645

OCRWM Program ´ t - Quality Assurance
Cost Pe .ce Report
by Functional Organization
Cumulative data thru March 31, 1998

Description		Month of March 1998					FY 96-FY 98					Project At Completion		
		Original Value of Work		Actual Cost of Work	Variance from Plan		Original Value of Work		Actual Cost of Work	Variance from Plan		Original Value of Work	Latest Revised Estimate	Variance from Plan
		Sched	Perf	Perf	Sched	Cost	Sched	Perf	Perf	Sched	Cost			
QA Programs HQ OWAST	9.1.1	65	65	74	0	-9	2630	2630	2377	0	253	3084	2905	179
YUCCA MOUNTAIN QUALITY ASSUR	9.1.2	695	695	710	0	-15	16356	16356	16100	0	256	20850	20594	256
AFFECTED ORGANIZATIONS	9.1.3	0	0	2	0	-2	18686	18686	17598	0	1088	18686	17615	1071
QA MANAGEMENT ASSESSMENT	9.1.4	35	35	43	0	-8	806	806	824	0	-18	1025	1043	-18
Project Baseline Budget - Authorized Work		795	795	829	0	-34	38478	38478	36899	0	1579	43645	42157	1488
HISTORICAL COSTS (PRE-FY1996)		0	0	0	0	0	0	0	0	0	0	0	0	0
BASELINE TO-COMPLETION		0	0	0	0	0	0	0	0	0	0	0	0	0
Total Project Baseline Budget		795	795	829	0	-34	38478	38478	36899	0	1579	43645	42157	1488
Undistributed Budget												0	0	0
Fee												830	830	0
Program Baseline Adjustment												0	0	
Performance Measurement Baseline		795	795	829	0	-34	38478	38478	36899	0	1579	44475	42987	1488
Management Reserve												473	473	0
Other												205	188	0
Total Project Cost												45153	43648	1505

OCRWM Program S[illegible]t - Quality Assurance
Cost P[illegible]e Report
by Functional Organization
Cumulative data thru March 31, 1998

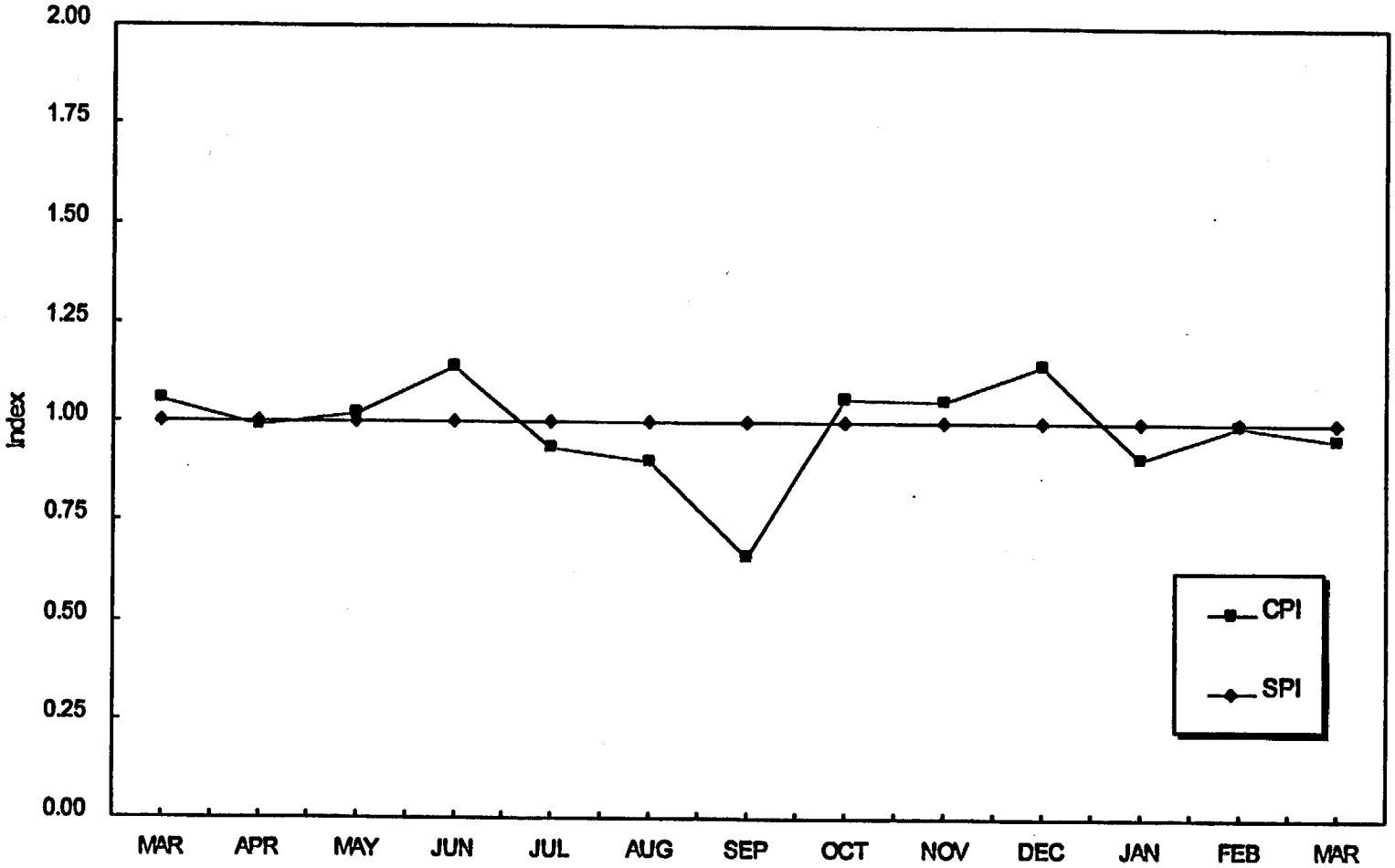
Description		Month of March 1998					FY 1998					Project at Completion FY 1998		
		Original Value of Work		Actual Cost of Work	Variance from Plan		Original Value of Work		Actual Cost of Work	Variance from Plan		Original Value of Work	Latest Revised Estimate	Variance from Plan
		Sched	Perf	Perf	Sched	Cost	Sched	Perf	Perf	Sched	Cost			
QA Programs HQ OWAST	9.1.1	65	65	74	0	-9	383	383	347	0	36	837	801	36
YUCCA MOUNTAIN QUALITY ASSUR	9.1.2	695	695	710	0	-15	3954	3954	3910	0	44	8448	8404	44
AFFECTED ORGANIZATIONS	9.1.3	0	0	2	0	-2	0	0	14	0	-14	0	31	-31
QA MANAGEMENT ASSESSMENT	9.1.4	35	35	43	0	-8	125	125	133	0	-8	344	352	-8
Project Baseline Budget - Authorized Work		795	795	829	0	-34	4462	4462	4404	0	58	9629	9588	41
HISTORICAL COSTS (PRE-FY1996)		0	0	0	0	0	0	0	0	0	0	0	0	0
BASELINE TO-COMPLETION		0	0	0	0	0	0	0	0	0	0	0	0	0
Total Project Baseline Budget		795	795	829	0	-34	4462	4462	4404	0	58	9629	9588	41
Undistributed Budget												0	0	0
Fee												0	0	0
Program Baseline Adjustment												0	0	
Performance Measurement Baseline		795	795	829	0	-34	4462	4462	4404	0	58	9629	9588	41
Management Reserve												423	423	0
Other												0	0	0
Total Project Cost												10052	10011	41

FY98
WBS 9.1 VARIANCE ANALYSIS
($000)

WBS	TITLE	BCWP	BCWS	ACWP	SV	CV
9.1	QUALITY ASSURANCE	4462	4462	4404	0	58

PERFORMANCE ANALYSIS FY 1997/1998

WBS 9.1, CPI & SPI OVERVIEW



U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Director's Program Review
Program Management and Administration

PRESENTED TO

Lake Barrett, Acting Director

PRESENTED BY

Sam Rousso, Director
Office of Program Management and Administration

May 7,1998

PRIORITY ACTIVITIES

- Complete TSLCC and Fee Adequacy Analyses
- Submit the Navy/RW MOA to Navy for Concurrence
- Issue CRD Revision 4 to Incorporate DOE SNF/ Hanford 15 ft Canister
- Revise the CRD to Include Surplus Plutonium Waste Forms
- Complete the Total System Description Revision
- Respond to the NWTRB 1997 Summary Report & Other TRB Letters

PRIORITY ACTIVITIES

- Prepare for June NRC Management Meeting
- Implement the Strategic System Management Policy
- Develop Fiscal Year 2000 Budget Request
- Develop Fiscal Year 1999 Annual Work Plans
- Complete FY2000-2004 OCRWM Planning Summary and FY2000 Performance Plan
- Obtain Secretarial Approval of Program Plan
- Obtain Secretarial Approval of Annual Report

PRIORITY ACTIVITIES

Information Resource Management

- Issue FY 1999 IM Multi-Year Program Plan
- Finalize Year 2000 Initiatives
- Brief and Implement Program-wide policy, procedures and technology for managing E-mail as a Record
- Implement a Capital Planning Process
- Upgrade the LAN

CHANGES SI CE LAST DPR
February-March

- WAD 98-50 Revision 3 Added $285,000 to the WBS 9.2.1 Contractor Budget Baseline to Accelerate Development of the TSLCC and Fee Adequacy Reports

ACCOMPLISHMENTS

March-April

- Systems Engineering
  - Established the DOE Nuclear Materials Disposition Task Team
  - Signed two new Annexes for Collaboration with Japan
  - Completed the Contingency Study/Briefed OMB and CFO

ACCOMPLISHMENTS

March-April

- Regulatory
  - Completed OCRWM Security Survey
  - Completed Sunset Review of RW Order, HQ 250.1, Exemption from DOE Directives
- Program Management
  - Approved the Strategic System Management Policy
  - Issued the FY 2000 Budget Development and FY 1999 Annual Work Plan Guidance

ACCOMPLISHMENTS

March-April

- Planning and Administration
  - Issued Spring 1998 OCRWM Enterprise
  - Launched Enhanced OCRWM Home Page
  - Submitted Year 2000 Cost Estimates to HR

ACCOMPLISHMENTS

March-April

- Contract Management
  - Transferred Management of the M&O Contract to Nevada Operations Office
- Human Resources
  - Implemented OCRWM's Reduction in Force

FY 1998 Major Deliverables

Description	Plan Date	Forecast/ Actual Date
OCRWM Issues Approved Strategic System Management Policy	01/02/97	04/8/98 A
Revise CRD to Include Surplus Plutonium Waste Forms BCP	03/31/98	05/29/98
Submit OCRWM Annual Report to Congress	04/15/98	06/15/98
Revise the CRWMS Total System Description	8/15/98	8/15/98
Submit FY 2000 OMB Request to CFO	09/03/98	09/03/98
TSLCC and Fee Adequacy Report to RW-1	11/30/98	09/15/98

DPR ACTION ITEMS

- RW-53 work with YMSCO to ensure that the VA milestones are at the proper level in our baselines
  - closed

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

DIRECTOR'S PROGRAM REVIEW

Program Management & Administration Performance Measurement

PRESENTED TO

LAKE BARRETT, Acting Director

OCRWM

PRESENTED BY

Vic Trebules, Director

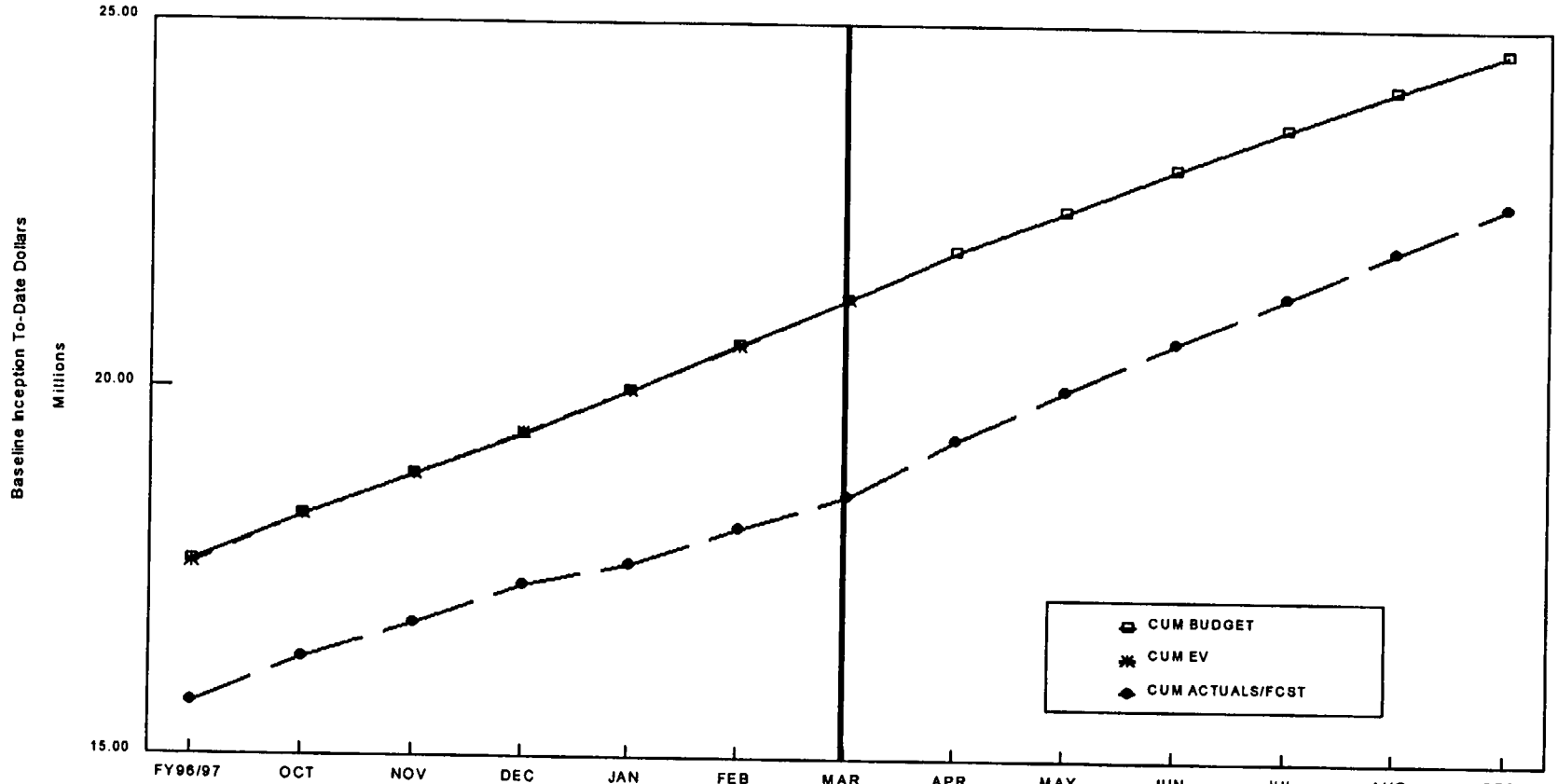
PM&A Program Management Division

May 7, 1998

COMPARISON OF PRIOR BUDGET BASELINE TO CURRENT BUDGET BASELINE ($K)

	WBS 9.2 PM&I	+	WBS 9.3 HR&A	=	PM&A
Baseline 9/30/97	17,676		22,469		40,145
Baseline Last DPR	24,417		30,788		55,205
Changes since Last DPR	+ 285		0	+	285
MAR (FY98)	24,702		30,788		55,490

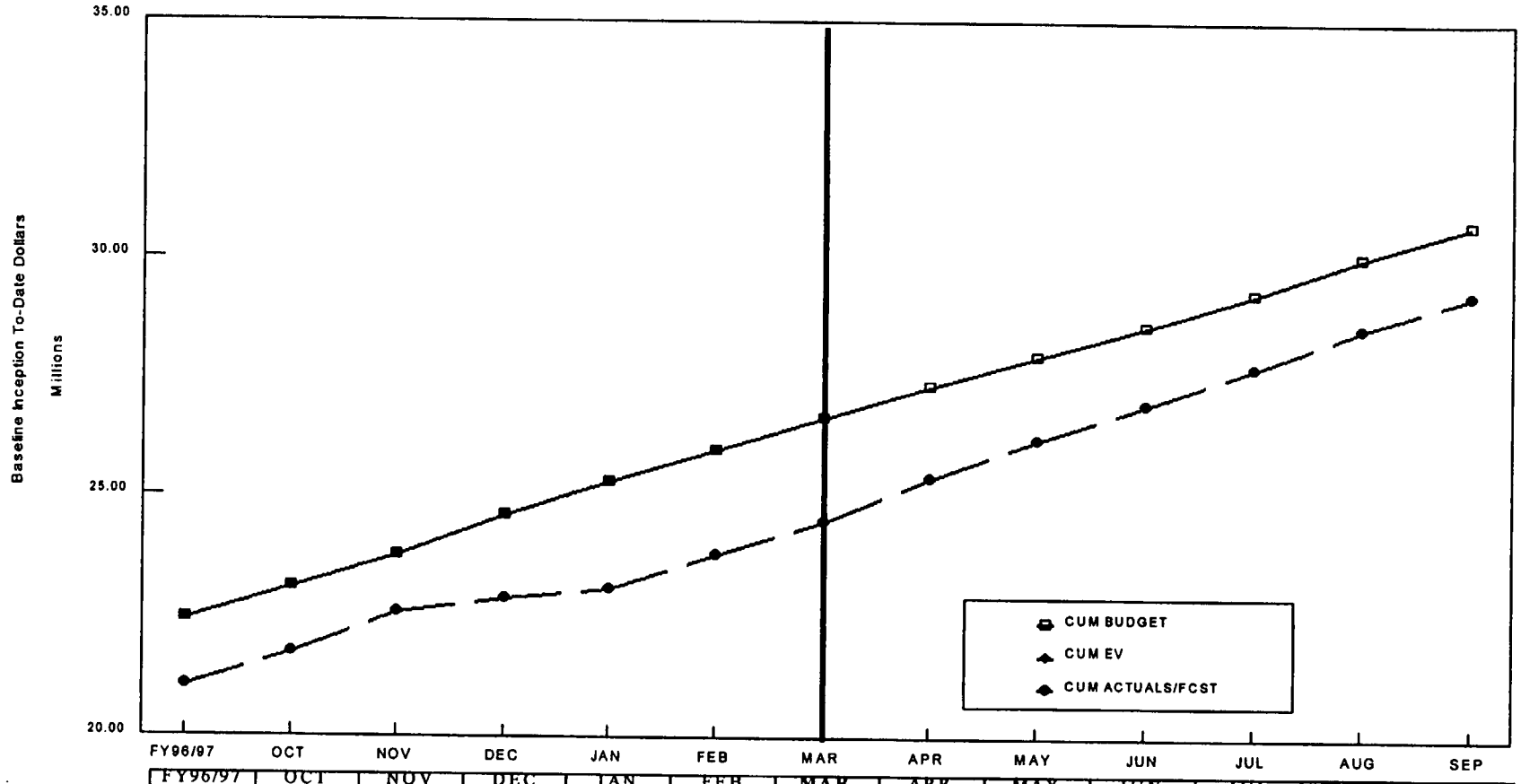
FY1996-98 9.2 PM&I FYTD EV Chart



	FY96/97	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP
MONTHLY BUD	17676	644	544	537	591	611	654	642	568	600	565	544	526
MONTHLY EV	17633	676	551	549	591	589	661						
MONTHLY ACT	15726	620	484	535	288	505	451						
MONTHLY FCST								748	671	662	642	632	610
SCHED VAR	-43	32	7	12	0	-22	7						
COST VAR	1907	56	67	14	303	84	210						
CUM BUDGET	17676	18320	18864	19401	19992	20603	21257	21899	22467	23067	23632	24176	24702
CUM EV	17633	18309	18860	19409	20000	20589	21250						
CUM ACTUAL	15726	16346	16830	17365	17653	18158	18609						
CUM FCST								19357	20028	20690	21332	21964	22574
ITD SV	-43	-11	-4	8	8	-14	-7						
ITD CV	1907	1963	2030	2044	2347	2431	2641						

5/6/98
DC9712PM

FY1996-98 9.3 HRCA FYTD EV Chart



	FY96/97	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP
MONTHLY BUD	22469	653	678	843	685	661	679	667	639	657	694	772	692
MONTHLY EV	22469	653	678	843	685	661	679						
MONTHLY ACT	21079	698	837	276	191	739	696						
MONTHLY FCST								893	799	764	770	854	719
SCHED VAR	0	0	0	0	0	0	0						
COST VAR	1390	-45	-159	567	494	-78	-17						
CUM BUDGET	22469	23121	23799	24642	25327	25988	26667	27334	27973	28630	29324	30096	30788
CUM EV	22469	23121	23799	24642	25327	25988	26667						
CUM ACTUAL	21079	21777	22614	22890	23081	23820	24516						
CUM FCST								25409	26208	26972	27742	28596	29315
ITD SV	0	0	0	0	0								
ITD CV	1390	1345	1186	1753	2247	2169	2152						

5/6/98
DC9712PM

FY98 PROJECT BUDGET PERFORMANCE
PM&A - March 1998

U.S Department of Energy
Cost Performance Report - Work Breakdown Structure (Format 1)

1. Title Program Support	2. Reporting Period 01-Mar-1998 to 31-Mar-1998	3. Dollars Expressed in Thousands
4. Participant Name & Address OCRWM	5. Cost Plan Date Apr-98	6. Report Date Apr-98
		7. Manager's Signature

8. WBS Elements	Current Period					FY98 to Date					FY98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
9.2 Program Management & Integration	654	661	498	7	163	3581	3556	2890	-25	666	7030	6798	232
9.3 Human Resources & Administration	679	679	698	0	-19	4199	4199	3649	0	550	8320	8346	-26
9. Contractor Budget Base	**1333**	**1340**	**1196**	**7**	**144**	**7780**	**7755**	**6539**	**-25**	**1216**	**15350**	**15144**	**206**
10. Federal Cost & Budget													
11. Undistributed Budget													
13. Fee													
14. Baseline Adjustment/Contingency											453	453	0
15. Perf. Measurement Baseline											**15803**	**15597**	**206**
16. Management Reserve													
17. Other													
18. Program Budget Base											15803	15597	206

5/6/98
DC9712PM

FY 96-98 PROJECT BUDGET PERFORMANCE
PM&A - March 1998

U.S Department of Energy
Cost Performance Report - Work Breakdown Structure (Format 1)

1. Title Program Support	2. Reporting Period 01-Mar-98 to 31-Mar-98	3. Dollars Expressed in Thousands Escalated
4. Participant Name & Address OCRWM	5. Cost Plan Date Apr-98	6. Report Date Apr-98
		7. Manager's Signature

8. WBS Elements	Current Period					FY96-98 to Date					FY96-98 At Completion of Work Year End		
	Budgeted Cost		Actual Cost of Work Perf	Variance		Budgeted Cost		Actual Cost of Work Perf	Variance				
Summary Level: 1 Detail Level: 2	Work Sched	Work Perf		Sched	Cost	Work Sched	Work Perf		Sched	Cost	Budget	Latest Revised Estimate	Var.
9.2 Program Management & Integration	654	661	451	7	210	21253	21250	18609	-3	2641	24702	22574	2128
9.3 Human Resources & Administration	679	679	696	0	-17	26667	26667	24524	0	2143	30788	29315	1473
9. Contractor Budget Base	**1333**	**1340**	**1147**	**7**	**193**	**47920**	**47917**	**43133**	**-3**	**4784**	**55490**	**51889**	**3601**
10. Federal Cost & Budget											**75647**	**72682**	**2965**
11. Undistributed Budget											0	0	0
13. Fee											3601	3385	216
14. Baseline Adjustment/Contingency											522	0	0
15. Perf. Measurement Baseline											**135260**	**127956**	**6782**
16. Management Reserve											3679	3679	0
17. Other											1391	1757	-366
18. Program Budget Base											140330	133392	6416

5/6/98
DC9712PM

PM&A VARIANCE ANALYSIS

($000)

WBS	TITLE	BCWS	BCWP	ACWP	SV	CV
9.2 & 9.3	PM&A	47920	47917	43133	-3	4784

Cost Variance:

The cost variance reflects FY96 and FY97 work completed below budget.

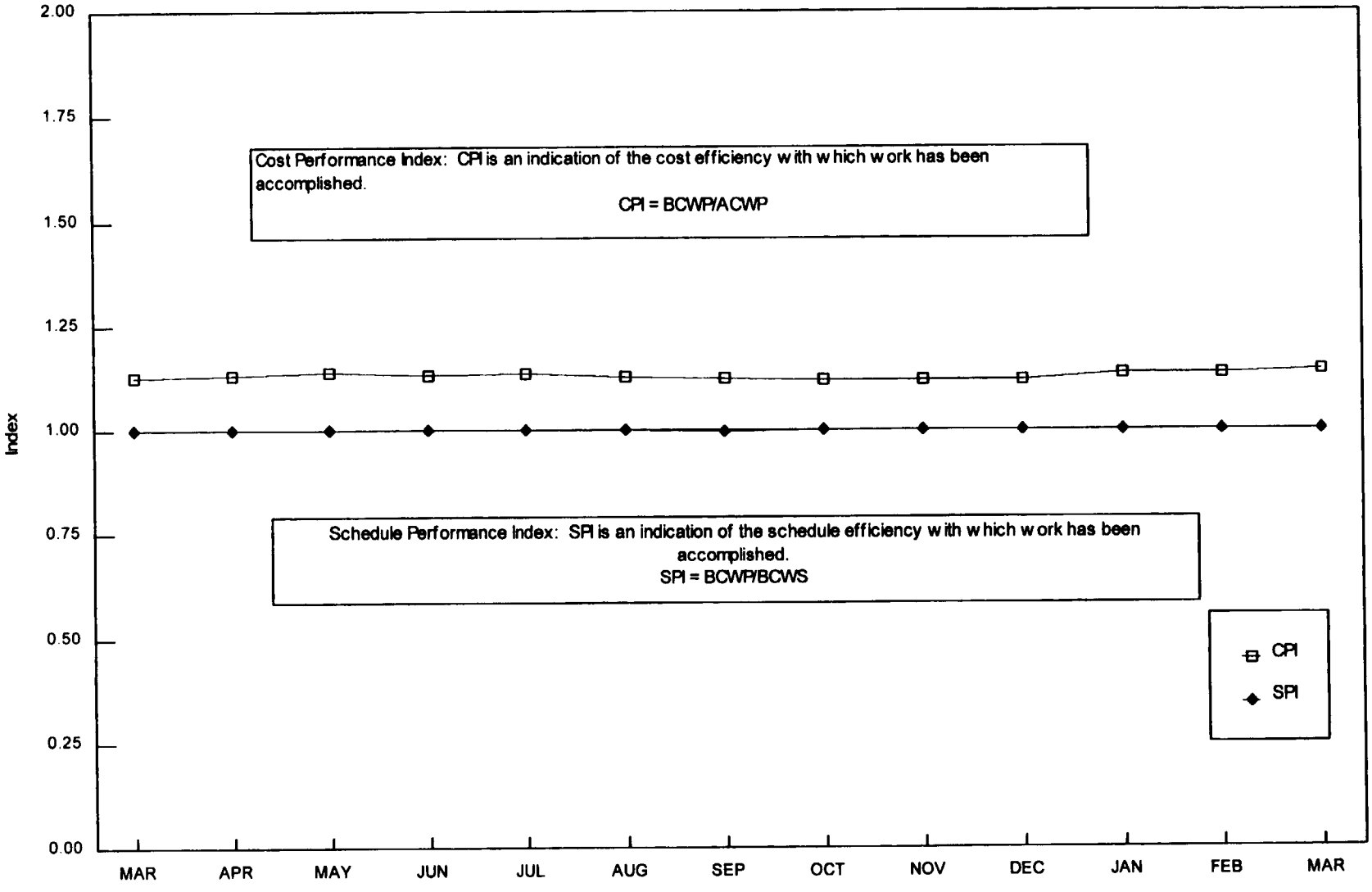
Impact:

- None

Corrective Action:

- None

FY 1997/98 WBS 9.2 CPI/SPI GRAPH



FY 97/98	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR
CPI	1.127	1.132	1.137	1.132	1.134	1.126	1.121	1.120	1.121	1.118	1.133	1.133	1.142
SPI	1.000	1.000	1.000	0.999	0.999	0.999	0.998	1.000	1.000	1.001	1.001	0.999	1.000

5/6/98
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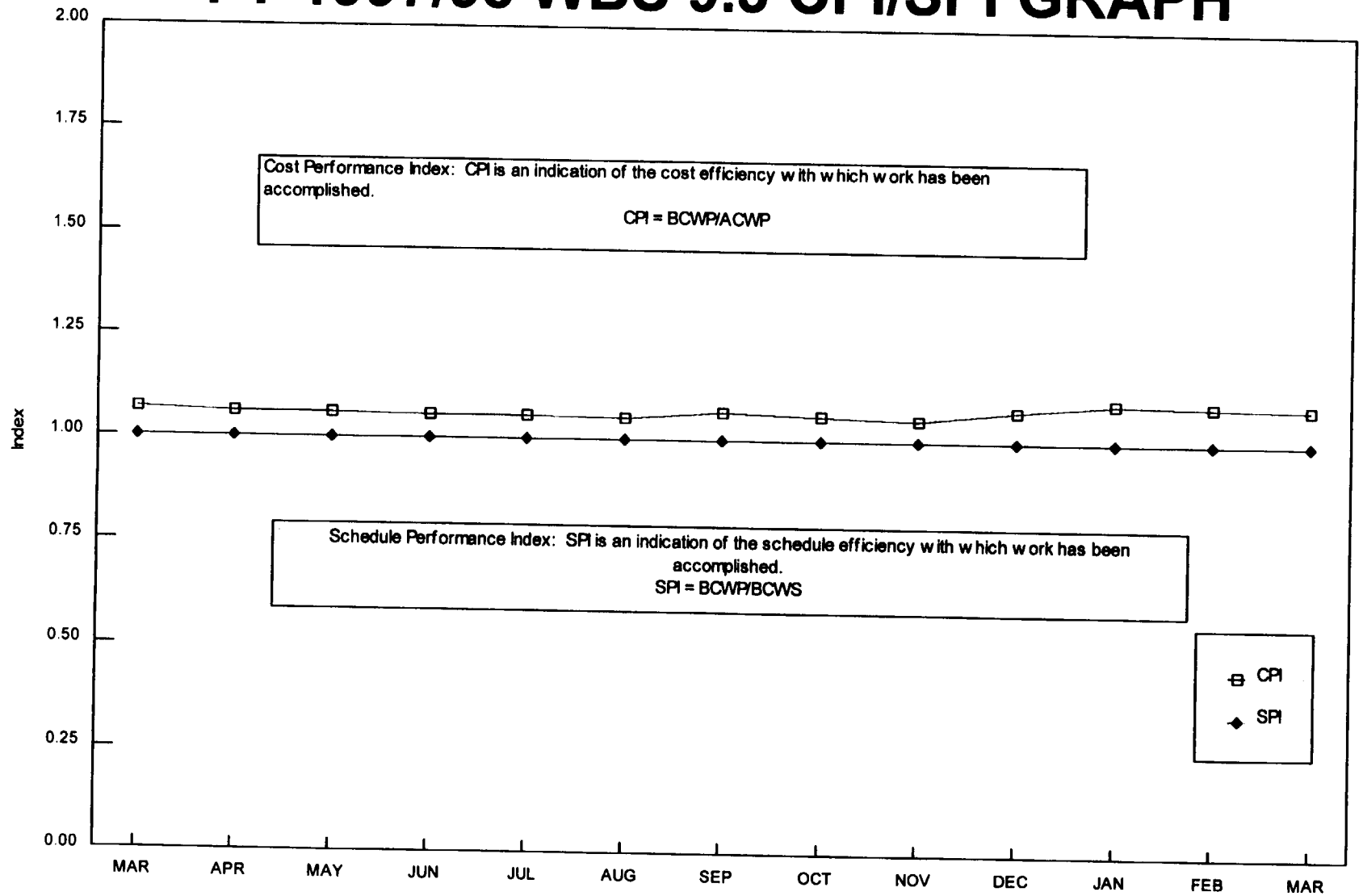
FY 1997/98 WBS 9.3 CPI/SPI GRAPH



FY 97/98	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR
CPI	1.066	1.061	1.061	1.056	1.055	1.053	1.066	1.062	1.052	1.077	1.097	1.091	1.087
SPI	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000

5/6/98
DC9712PM